50.92, 50.100, 50.101, 50.102, 50.103, 50.109, 50.110, 50.111.

[57 FR 55075, Nov. 24, 1992, as amended at 61 FR 39303, July 29, 1996]

§ 50.120 Training and qualification of nuclear power plant personnel.

(a) *Applicability*. The requirements of this section apply to each applicant for (applicant) and each holder of an operating license (licensee) for a nuclear power plant of the type specified in § 50.21(b) or § 50.22.

(b) *Requirements*. (1) Each nuclear power plant applicant, by November 22, 1993 or 18 months prior to fuel load, whichever is later, and each nuclear power plant licensee, by November 22, 1993 shall establish, implement, and maintain a training program derived from a systems approach to training as defined in 10 CFR 55.4. The training program must provide for the training and qualification of the following categories of nuclear power plant personnel:

(i) Non-licensed operator.

(ii) Shift supervisor.

(iii) Shift technical advisor.

(iv) Instrument and control technician.

(v) Electrical maintenance personnel.

(vi) Mechanical maintenance personnel.

(vii) Radiological protection technician.

(viii) Chemistry technician.

(ix) Engineering support personnel.

(2) The training program must incorporate the instructional requirements necessary to provide qualified personnel to operate and maintain the facility in a safe manner in all modes of operation. The training program must be developed so as to be in compliance with the facility license, including all technical specifications and applicable regulations. The training program must be periodically evaluated and revised as appropriate to reflect industry experience as well as changes to the facility, procedures, regulations, and quality assurance requirements. The training program must be periodically reviewed by licensee management for effectiveness. Sufficient records must be maintained by the licensee to maintain program integrity and kept available for NRC inspection to verify the adequacy of the program.

[58 FR 21912, Apr. 26, 1993; 58 FR 39092, July 21, 1993]

APPENDIX A TO PART 50—GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

Table of Contents

CRITERIA—Continued

	Number
Inspection of Emergency Core Cooling System	36
Testing of Emergency Core Cooling System	37
Containment Heat Removal	38
Inspection of Containment Heat Removal System	39
Testing of Containment Heat Removal System	40
Containment Atmosphere Cleanup	41
Inspection of Containment Atmosphere Cleanup Systems	42
Testing of Containment Atmosphere Cleanup Systems	43
Cooling Water	44
Inspection of Cooling Water System	45
Testing of Cooling Water System	46
V. *Reactor Containment:*	
Containment Design Basis	50
Fracture Prevention of Containment Pressure Boundary	51
Capability for Containment Leakage Rate Testing	52
Provisions for Containment Testing and Inspection	53
Systems Penetrating Containment	54
Reactor Coolant Pressure Boundary Penetrating Containment	55
Primary Containment Isolation	56
Closed Systems Isolation Valves	57
VI. *Fuel and Radioactivity Control:*	
Control of Releases of Radioactive Materials to the Environment	60
Fuel Storage and Handling and Radioactivity Control	61
Prevention of Criticality in Fuel Storage and Handling	62
Monitoring Fuel and Waste Storage	63
Monitoring Radioactivity Releases	64

INTRODUCTION

Pursuant to the provisions of §50.34, an application for a construction permit must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

The development of these General Design Criteria is not yet complete. For example, some of the definitions need further amplification. Also, some of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined. Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include:

(1) Consideration of the need to design against single failures of passive components in fluid systems important to safety. (See Definition of Single Failure.)

(2) Consideration of redundancy and diversity requirements for fluid systems important to safety. A ''system'' could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem, and the required interconnection and independence of the subsystems have not yet been developed or defined. (See Criteria 34, 35, 38, 41, and 44.)

(3) Consideration of the type, size, and orientation of possible breaks in components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss-of-coolant accidents. (See Definition of Loss of Coolant Accidents.)

(4) Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of protection systems and reactivity control systems. (See Criteria 22, 24, 26, and 29.)

It is expected that the criteria will be augmented and changed from time to time as important new requirements for these and other features are developed.

There will be some water-cooled nuclear power plants for which the General Design Criteria are not sufficient and for which additional criteria must be identified and satisfied in the interest of public safety. In particular, it is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

DEFINITIONS AND EXPLANATIONS

Nuclear power unit. A nuclear power unit means a nuclear power reactor and associated equipment necessary for electric power generation and includes those structures, systems, and components required to provide reasonable assurance the facility can be operated without undue risk to the health and safety of the public.

Loss of coolant accidents. Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.[1]

Single failure. A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions.[2]

Anticipated operational occurrences. Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

CRITERIA

I. Overall Requirements

Criterion 1—Quality standards and records. Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Criterion 2—Design bases for protection against natural phenomena. Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Criterion 3—Fire protection. Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Criterion 4—Environmental and dynamic effects design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

[1] Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development.

[2] Single failures of passive components in electric systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.

Criterion 5—Sharing of structures, systems, and components. Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

II. Protection by Multiple Fission Product Barriers

Criterion 10—Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Criterion 11—Reactor inherent protection. The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Criterion 12—Suppression of reactor power oscillations. The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Criterion 13—Instrumentation and control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Criterion 14—Reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Criterion 15—Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Criterion 16—Containment design. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Criterion 17—Electric power systems. An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

Criterion 18—Inspection and testing of electric power systems. Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Criterion 19—Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses under part 52 of this chapter who do not reference a standard design certification, or holders of operating licenses using an alternative source term under §50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in §50.2 for the duration of the accident.

III. Protection and Reactivity Control Systems

Criterion 20—Protection system functions. The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Criterion 21—Protection system reliability and testability. The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Criterion 22—Protection system independence. The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Criterion 23—Protection system failure modes. The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Criterion 24—Separation of protection and control systems. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Criterion 25—Protection system requirements for reactivity control malfunctions. The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Criterion 26—Reactivity control system redundancy and capability. Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Criterion 27—Combined reactivity control systems capability. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Criterion 28—Reactivity limits. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Criterion 29—Protection against anticipated operational occurrences. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

IV. Fluid Systems

Criterion 30—Quality of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Criterion 31—Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Criterion 32—Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Criterion 33—Reactor coolant makeup. A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Criterion 34—Residual heat removal. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 35—Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 36—Inspection of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Criterion 37—Testing of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 38—Containment heat removal. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 39—Inspection of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Criterion 40—Testing of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 41—Containment atmosphere cleanup. Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Criterion 42—Inspection of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Criterion 43—Testing of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Criterion 44—Cooling water. A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 45—Inspection of cooling water system. The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

Criterion 46—Testing of cooling water system. The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

V. Reactor Containment

Criterion 50—Containment design basis. The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by §50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Criterion 51—Fracture prevention of containment pressure boundary. The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.

Criterion 52—Capability for containment leakage rate testing. The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Criterion 53—Provisions for containment testing and inspection. The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

Criterion 54—Piping systems penetrating containment. Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Criterion 55—Reactor coolant pressure boundary penetrating containment. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Criterion 56—Primary containment isolation. Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Criterion 57—Closed system isolation valves. Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

VI. Fuel and Radioactivity Control

Criterion 60—Control of releases of radioactive materials to the environment. The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Criterion 61—Fuel storage and handling and radioactivity control. The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Criterion 62—Prevention of criticality in fuel storage and handling. Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Criterion 63—Monitoring fuel and waste storage. Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Criterion 64—Monitoring radioactivity releases. Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

[36 FR 3256, Feb. 20, 1971, as amended at 36 FR 12733, July 7, 1971; 41 FR 6258, Feb. 12, 1976; 43 FR 50163, Oct. 27, 1978; 51 FR 12505, Apr. 11, 1986; 52 FR 41294, Oct. 27, 1987; 64 FR 72002, Dec. 23, 1999]

APPENDIX B TO PART 50—QUALITY ASSURANCE CRITERIA FOR NUCLEAR POWER PLANTS AND FUEL REPROCESSING PLANTS

Introduction. Every applicant for a construction permit is required by the provisions of §50.34 to include in its preliminary safety analysis report a description of the quality assurance program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility. Every applicant for an operating license is required to include, in its final safety analysis report, information pertaining to the managerial and administrative controls to be used to assure safe operation. Nuclear power plants and fuel reprocessing plants include structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. This appendix establishes quality assurance requirements for the design, construction, and operation of those structures, systems, and components. The pertinent requirements of this appendix apply to all activities affecting the safety-related functions of those structures, systems, and components; these activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying.

As used in this appendix, ''quality assurance'' comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system which provide a means to control the quality of the material, structure, component, or system to predetermined requirements.

I. ORGANIZATION

The applicant[1] shall be responsible for the establishment and execution of the quality assurance program. The applicant may delegate to others, such as contractors, agents, or consultants, the work of establishing and executing the quality assurance program, or any part thereof, but shall retain responsibility therefor. The authority and duties of persons and organizations performing activities affecting the safety-related functions of structures, systems, and components shall be clearly established and delineated in writing. These activities include both the performing functions of attaining quality objectives and the quality assurance functions. The quality assurance functions are those of (a) assuring that an appropriate quality assurance program is established and effectively executed and (b) verifying, such as by checking, auditing, and inspection, that activities affecting the safety-related functions have been correctly performed. The persons and organizations performing quality assurance functions shall have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions. Such persons and organizations performing quality assurance functions shall report to a management level such that this required authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations, are provided. Because of the many variables involved, such as the number of personnel, the type of activity being performed, and the location or locations where activities are performed, the organizational structure for executing the quality assurance program may take various forms provided that the persons and organizations assigned the quality assurance functions have this required authority and organizational freedom. Irrespective of the organizational structure, the individual(s) assigned the responsibility for assuring effective execution of any portion of the quality assurance program at any location where activities subject to this appendix are being performed shall have direct access to such levels of management as may be necessary to perform this function.

II. QUALITY ASSURANCE PROGRAM

The applicant shall establish at the earliest practicable time, consistent with the schedule for accomplishing the activities, a quality assurance program which complies with the requirements of this appendix. This program shall be documented by written policies, procedures, or instructions and shall be carried out throughout plant life in accordance with those policies, procedures, or instructions. The applicant shall identify the structures, systems, and components to be covered by the quality assurance program and the major organizations participating in the program, together with the designated functions of these organizations. The quality assurance program shall provide control over activities affecting the quality of the identified structures, systems, and components, to

[1] While the term ''applicant'' is used in these criteria, the requirements are, of course, applicable after such a person has received a license to construct and operate a nuclear power plant or a fuel reprocessing plant. These criteria will also be used for guidance in evaluating the adequacy of quality assurance programs in use by holders of construction permits and operating licenses.

an extent consistent with their importance to safety. Activities affecting quality shall be accomplished under suitably controlled conditions. Controlled conditions include the use of appropriate equipment; suitable environmental conditions for accomplishing the activity, such as adequate cleanness; and assurance that all prerequisites for the given activity have been satisfied. The program shall take into account the need for special controls, processes, test equipment, tools, and skills to attain the required quality, and the need for verification of quality by inspection and test. The program shall provide for indoctrination and training of personnel performing activities affecting quality as necessary to assure that suitable proficiency is achieved and maintained. The applicant shall regularly review the status and adequacy of the quality assurance program. Management of other organizations participating in the quality assurance program shall regularly review the status and adequacy of that part of the quality assurance program which they are executing.

III. DESIGN CONTROL

Measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in §50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components.

Measures shall be established for the identification and control of design interfaces and for coordination among participating design organizations. These measures shall include the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces.

The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. The verifying or checking process shall be performed by individuals or groups other than those who performed the original design, but who may be from the same organization. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, it shall include suitable qualifications testing of a prototype unit under the most adverse design conditions. Design control measures shall be applied to items such as the following: reactor physics, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests.

Design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design and be approved by the organization that performed the original design unless the applicant designates another responsible organization.

IV. PROCUREMENT DOCUMENT CONTROL

Measures shall be established to assure that applicable regulatory requirements, design bases, and other requirements which are necessary to assure adequate quality are suitably included or referenced in the documents for procurement of material, equipment, and services, whether purchased by the applicant or by its contractors or subcontractors. To the extent necessary, procurement documents shall require contractors or subcontractors to provide a quality assurance program consistent with the pertinent provisions of this appendix.

V. INSTRUCTIONS, PROCEDURES, AND DRAWINGS

Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

VI. DOCUMENT CONTROL

Measures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality. These measures shall assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel and are distributed to and used at the location where the prescribed activity is performed. Changes to documents shall be reviewed and approved by the same organizations that performed the original review and approval unless the applicant designates another responsible organization.

VII. CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

Measures shall be established to assure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents. These measures shall include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor or subcontractor, inspection at the contractor or subcontractor source, and examination of products upon delivery. Documentary evidence that material and equipment conform to the procurement requirements shall be available at the nuclear power plant or fuel reprocessing plant site prior to installation or use of such material and equipment. This documentary evidence shall be retained at the nuclear power plant or fuel reprocessing plant site and shall be sufficient to identify the specific requirements, such as codes, standards, or specifications, met by the purchased material and equipment. The effectiveness of the control of quality by contractors and subcontractors shall be assessed by the applicant or designee at intervals consistent with the importance, complexity, and quantity of the product or services.

VIII. IDENTIFICATION AND CONTROL OF MATERIALS, PARTS, AND COMPONENTS

Measures shall be established for the identification and control of materials, parts, and components, including partially fabricated assemblies. These measures shall assure that identification of the item is maintained by heat number, part number, serial number, or other appropriate means, either on the item or on records traceable to the item, as required throughout fabrication, erection, installation, and use of the item. These identification and control measures shall be designed to prevent the use of incorrect or defective material, parts, and components.

IX. CONTROL OF SPECIAL PROCESSES

Measures shall be established to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

X. INSPECTION

A program for inspection of activities affecting quality shall be established and executed by or for the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. Such inspection shall be performed by individuals other than those who performed the activity being inspected. Examinations, measurements, or tests of material or products processed shall be performed for each work operation where necessary to assure quality. If inspection of processed material or products is impossible or disadvantageous, indirect control by monitoring processing methods, equipment, and personnel shall be provided. Both inspection and process monitoring shall be provided when control is inadequate without both. If mandatory inspection hold points, which require witnessing or inspecting by the applicant's designated representative and beyond which work shall not proceed without the consent of its designated representative are required, the specific hold points shall be indicated in appropriate documents.

XI. TEST CONTROL

A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. The test program shall include, as appropriate, proof tests prior to installation, preoperational tests, and operational tests during nuclear power plant or fuel reprocessing plant operation, of structures, systems, and components. Test procedures shall include provisions for assuring that all prerequisites for the given test have been met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. Test results shall be documented and evaluated to assure that test requirements have been satisfied.

XII. CONTROL OF MEASURING AND TEST EQUIPMENT

Measures shall be established to assure that tools, gages, instruments, and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specified periods to maintain accuracy within necessary limits.

XIII. HANDLING, STORAGE AND SHIPPING

Measures shall be established to control the handling, storage, shipping, cleaning and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration. When necessary for particular products, special protective environments, such as inert gas atmosphere, specific moisture content levels, and temperature levels, shall be specified and provided.

XIV. INSPECTION, TEST, AND OPERATING STATUS

Measures shall be established to indicate, by the use of markings such as stamps, tags, labels, routing cards, or other suitable means, the status of inspections and tests performed upon individual items of the nuclear power plant or fuel reprocessing plant. These measures shall provide for the identification of items which have satisfactorily passed required inspections and tests, where necessary to preclude inadvertent bypassing of such inspections and tests. Measures shall also be established for indicating the operating status of structures, systems, and components of the nuclear power plant or fuel reprocessing plant, such as by tagging valves and switches, to prevent inadvertent operation.

XV. NONCONFORMING MATERIALS, PARTS, OR COMPONENTS

Measures shall be established to control materials, parts, or components which do not conform to requirements in order to prevent their inadvertent use or installation. These measures shall include, as appropriate, procedures for identification, documentation, segregation, disposition, and notification to affected organizations. Nonconforming items shall be reviewed and accepted, rejected, repaired or reworked in accordance with documented procedures.

XVI. CORRECTIVE ACTION

Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management.

XVII. QUALITY ASSURANCE RECORDS

Sufficient records shall be maintained to furnish evidence of activities affecting quality. The records shall include at least the following: Operating logs and the results of reviews, inspections, tests, audits, monitoring of work performance, and materials analyses. The records shall also include closely-related data such as qualifications of personnel, procedures, and equipment. Inspection and test records shall, as a minimum, identify the inspector or data recorder, the type of observation, the results, the acceptability, and the action taken in connection with any deficiencies noted. Records shall be identifiable and retrievable. Consistent with applicable regulatory requirements, the applicant shall establish requirements concerning record retention, such as duration, location, and assigned responsibility.

XVIII. AUDITS

A comprehensive system of planned and periodic audits shall be carried out to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. The audits shall be performed in accordance with the written procedures or check lists by appropriately trained personnel not having direct responsibilities in the areas being audited. Audit results shall be documented and reviewed by management having responsibility in the area audited. Followup action, including reaudit of deficient areas, shall be taken where indicated.

[35 FR 10499, June 27, 1970, as amended at 36 FR 18301, Sept. 11, 1971; 40 FR 3210D, Jan. 20, 1975]

APPENDIX C TO PART 50—A GUIDE FOR THE FINANCIAL DATA AND RELATED INFORMATION REQUIRED TO ESTABLISH FINANCIAL QUALIFICATIONS FOR FACILITY CONSTRUCTION PERMITS

GENERAL INFORMATION

This appendix is intended to apprise applicants for licenses to construct production or utilization facilities of the types described in §50.21(b) or §50.22, or testing facilities, of the general kinds of financial data and other related information that will demonstrate the financial qualification of the applicant to carry out the activities for which the permit is sought. The kind and depth of information described in this guide is not intended to be a rigid and absolute requirement. In some instances, additional pertinent material may be needed. In any case, the applicant should include information other than that specified, if such information is pertinent to establishing the applicant's financial ability to construct the proposed facility.

It is important to observe also that both §50.33(f) and this appendix distinguish between applicants which are established organizations and those which are newly-formed entities organized primarily for the purpose of engaging in the activity for which the permit is sought. Those in the former category will normally have a history of operating experience and be able to submit financial statements reflecting the financial results of past operations. With respect, however, to the applicant which is a newly formed company established primarily for the purpose of carrying out the licensed activity, with little or no prior operating history, somewhat more detailed data and supporting documentation will generally be necessary. For this reason, the appendix describes separately the scope of information to be included in applications by each of these two classes of applicants.

In determining an applicant's financial qualification, the Commission will require the minimum amount of information necessary for that purpose. No special forms are prescribed for submitting the information. In many cases, the financial information usually contained in current annual financial reports, including summary data of prior years, will be sufficient for the Commission's needs. The Commission reserves the right, however, to require additional financial information at the construction permit stage, particularly in cases in which the proposed power generating facility will be commonly owned by two or more existing companies or in which financing depends upon long-term arrangements for sharing of the power from the facility by two or more electrical generating companies.

Applicants are encouraged to consult with the Commission with respect to any questions they may have relating to the requirements of the Commission's regulations or the information set forth in this appendix.

I. Applicants Which Are Established Organizations

A. Applications for construction permits

1. *Estimate of construction costs.* For electric utilities, each applicant's estimate of the total cost of the proposed facility should be broken down as follows and be accompanied by a statement describing the bases from which the estimate is derived:

(a) Total nuclear production plant costs	$..........
(b) Transmission, distribution, and general plant costs	$..........
(c) Nuclear fuel inventory cost for first core [1]	$..........
Total estimated cost	$..........

[1] Section 2.790 of 10 CFR part 2 and § 9.5 of 10 CFR part 9 indicate the circumstances under which information submitted by applicants may be withheld from public disclosure.

If the fuel is to be acquired by lease or other arrangement than purchase, the application should so state. The items to be included in these categories should be the same as those defined in the applicable electric plant and nuclear fuel inventory accounts prescribed by the Federal Energy Regulatory Commission or an explanation given as to any departure therefrom.

Since the composition of construction cost estimates for production and utilization facilities other than nuclear power reactors will vary according to the type of facility, no particular format is suggested for submitting such estimates. The estimate should, however, be itemized by categories of cost in sufficient detail to permit an evaluation of its reasonableness.

2. *Source of construction funds.* The application should include a brief statement of the applicant's general financial plan for financing the cost of the facility, identifying the source or sources upon which the applicant relies for the necessary construction funds, e.g., internal sources such as undistributed earnings and depreciation accruals, or external sources such as borrowings.

3. *Applicant's financial statements.* The application should also include the applicant's latest published annual financial report, together with any current interim financial statements that are pertinent. If an annual financial report is not published, the balance sheet and operating statement covering the latest complete accounting year together with all pertinent notes thereto and certification by a public accountant should be furnished.

II. Applicants Which Are Newly Formed Entities

A. Applications for construction permits

1. *Estimate of construction costs.* The information that will normally be required of applicants which are newly formed entities will not differ in scope from that required of established organizations. Accordingly, applicants should submit estimates as described above for established organizations.

2. *Source of construction funds.* The application should specifically identify the source or sources upon which the applicant relies for the funds necessary to pay the cost of constructing the facility, and the amount to be obtained from each. With respect to each source, the application should describe in detail the applicant's legal and financial relationships with its stockholders, corporate affiliates, or others (such as financial institutions) upon which the applicant is relying for financial assistance. If the sources of funds relied upon include parent companies or other corporate affiliates, information to support the financial capability of each such company or affiliate to meet its commitments to the applicant should be set forth in the application. This information should be of the same kind and scope as would be required if the parent companies or affiliates were in fact the applicant. Ordinarily, it will be necessary that copies of agreements or contracts among the companies be submitted.

As noted earlier in this appendix, an applicant which is a newly formed entity will normally not be in a position to submit the usual types of balance sheets and income statements reflecting the results of prior operations. The applicant should, however, include in its application a statement of its assets, liabilities, and capital structure as of the date of the application.

III. ANNUAL FINANCIAL STATEMENT

Each holder of a construction permit for a production or utilization facility of a type described in §50.21(b) or §50.22, or a testing facility is required by §50.71(b) to file its annual financial report with the Commission at the time of issuance thereof. This requirement does not apply to licensees or holders of construction permits for medical and research reactors.

IV. ADDITIONAL INFORMATION

The Commission may, from time to time, request the applicant, whether an established organization or newly formed entity, to submit additional or more detailed information respecting its financial arrangements and status of funds if such information is deemed necessary to enable the Commiasion to determine an applicant's financial qualifications for the license.

[49 FR 35753, Sept. 12, 1984, as amended at 50 FR 18853, May 3, 1985]

APPENDIX D TO PART 50 [RESERVED]

APPENDIX E TO PART 50—EMERGENCY PLANNING AND PREPAREDNESS FOR PRODUCTION AND UTILIZATION FACILITIES

Table of Contents

I. INTRODUCTION

Each applicant for a construction permit is required by §50.34(a) to include in the preliminary safety analysis report a discussion of preliminary plans for coping with emergencies. Each applicant for an operating license is required by §50.34(b) to include in the final safety analysis report plans for coping with emergencies.

This appendix establishes minimum requirements for emergency plans for use in attaining an acceptable state of emergency preparedness. These plans shall be described generally in the preliminary safety analysis report and submitted as part of the final safety analysis report.

The potential radiological hazards to the public associated with the operation of research and test reactors and fuel facilities licensed under 10 CFR parts 50 and 70 involve considerations different than those associated with nuclear power reactors. Consequently, the size of Emergency Planning Zones [1] (EPZs) for facilities other than power reactors and the degree to which compliance with the requirements of this section and sections II, III, IV, and V as necessary will be determined on a case-by-case basis.[2]

Notwithstanding the above paragraphs, in the case of an operating license authorizing only fuel loading and/or low power operations up to 5% of rated power, no NRC or FEMA review, findings, or determinations concerning the state of offsite emergency preparedness or the adequacy of and the capability to implement State and local offsite emergency plans, as defined in this appendix, are required prior to the issuance of such a license.

II. THE PRELIMINARY SAFETY ANALYSIS REPORT

The Preliminary Safety Analysis Report shall contain sufficient information to ensure the compatibility of proposed emergency plans for both onsite areas and the EPZs, with facility design features, site layout, and site location with respect to such considerations as access routes, surrounding population distributions, land use, and local jurisdictional boundaries for the EPZs in the case of nuclear power reactors as well as the means by which the standards of §50.47(b) will be met.

As a minimum, the following items shall be described:

A. Onsite and offsite organizations for coping with emergencies and the means for notification, in the event of an emergency, of persons assigned to the emergency organizations.

[1] EPZs for power reactors are discussed in NUREG–0396; EPA 520/1–78–016, ''Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants,'' December 1978. The size of the EPZs for a nuclear power plant shall be determined in relation to local emergency response needs and capabilities as they are affected by such conditions as demography, topography, land characteristics, access routes, and jurisdictional boundaries. The size of the EPZs also may be determined on a case-by-case basis for gas-cooled nuclear reactors and for reactors with an authorized power level less than 250 MW thermal. Generally, the plume exposure pathway EPZ for nuclear power plants with an authorized power level greater than 250 MW thermal shall consist of an area about 10 miles (16 km) in radius and the ingestion pathway EPZ shall consist of an area about 50 miles (80 km) in radius.

[2] Regulatory Guide 2.6 will be used as guidance for the acceptability of research and test reactor emergency response plans.

B. Contacts and arrangements made and documented with local, State, and Federal governmental agencies with responsibility for coping with emergencies, including identification of the principal agencies.

C. Protective measures to be taken within the site boundary and within each EPZ to protect health and safety in the event of an accident; procedures by which these measures are to be carried out (e.g., in the case of an evacuation, who authorizes the evacuation, how the public is to be notified and instructed, how the evacuation is to be carried out); and the expected response of offsite agencies in the event of an emergency.

D. Features of the facility to be provided for onsite emergency first aid and decontamination and for emergency transportation of onsite individuals to offsite treatment facilities.

E. Provisions to be made for emergency treatment at offsite facilities of individuals injured as a result of licensed activities.

F. Provisions for a training program for employees of the licensee, including those who are assigned specific authority and responsibility in the event of an emergency, and for other persons who are not employees of the licensee but whose assistance may be needed in the event of a radiological emergency.

G. A preliminary analysis that projects the time and means to be employed in the notification of State and local governments and the public in the event of an emergency. A nuclear power plant applicant shall perform a preliminary analysis of the time required to evacuate various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations, noting major impediments to the evacuation or taking of protective actions.

H. A preliminary analysis reflecting the need to include facilities, systems, and methods for identifying the degree of seriousness and potential scope of radiological consequences of emergency situations within and outside the site boundary, including capabilities for dose projection using real-time meteorological information and for dispatch of radiological monitoring teams within the EPZs; and a preliminary analysis reflecting the role of the onsite technical support center and of the near-site emergency operations facility in assessing information, recommending protective action, and disseminating information to the public.

III. The Final Safety Analysis Report

The Final Safety Analysis Report shall contain the plans for coping with emergencies. The plans shall be an expression of the overall concept of operation; they shall describe the essential elements of advance planning that have been considered and the provisions that have been made to cope with emergency situations. The plans shall incorporate information about the emergency response roles of supporting organizations and offsite agencies. That information shall be sufficient to provide assurance of coordination among the supporting groups and with the licensee.

The plans submitted must include a description of the elements set out in section IV for the Emergency Planning Zones (EPZs) to an extent sufficient to demonstrate that the plans provide reasonable assurance that adequate protective measures can and will be taken in the event of an emergency.

IV. Content of Emergency Plans

The applicant's emergency plans shall contain, but not necessarily be limited to, information needed to demonstrate compliance with the elements set forth below, i.e., organization for coping with radiation emergencies, assessment action, activation of emergency organization, notification procedures, emergency facilities and equipment, training, maintaining emergency preparedness, and recovery. In addition, the emergency response plans submitted by an applicant for a nuclear power reactor operating license shall contain information needed to demonstrate compliance with the standards described in §50.47(b), and they will be evaluated against those standards. The nuclear power reactor operating license applicant shall also provide an analysis of the time required to evacuate and for taking other protective actions for various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations.

A. Organization

The organization for coping with radiological emergencies shall be described, including definition of authorities, responsibilities, and duties of individuals assigned to the licensee's emergency organization and the means for notification of such individuals in the event of an emergency. Specifically, the following shall be included:

1. A description of the normal plant operating organization.

2. A description of the onsite emergency response organization with a detailed discussion of:

a. Authorities, responsibilities, and duties of the individual(s) who will take charge during an emergency;

b. Plant staff emergency assignments;

c. Authorities, responsibilities, and duties on an onsite emergency coordinator who shall be in charge of the exchange of information with offsite authorities responsible for coordinating and implementing offsite emergency measures.

3. A description, by position and function to be performed, of the licensee's headquarters personnel who will be sent to the

plant site to augment the onsite emergency organization.

4. Identification, by position and function to be performed, of persons within the licensee organization who will be responsible for making offsite dose projections, and a description of how these projections will be made and the results transmitted to State and local authorities, NRC, and other appropriate governmental entities.

5. Identification, by position and function to be performed, of other employees of the licensee with special qualifications for coping with emergency conditions that may arise. Other persons with special qualifications, such as consultants, who are not employees of the licensee and who may be called upon for assistance for emergencies shall also be identified. The special qualifications of these persons shall be described.

6. A description of the local offsite services to be provided in support of the licensee's emergency organization.

7. Identification of, and assistance expected from, appropriate State, local, and Federal agencies with responsibilities for coping with emergencies.

8. Identification of the State and/or local officials responsible for planning for, ordering, and controlling appropriate protective actions, including evacuations when necessary.

B. Assessment Actions

The means to be used for determining the magnitude of, and for continually assessing the impact of, the release of radioactive materials shall be described, including emergency action levels that are to be used as criteria for determining the need for notification and participation of local and State agencies, the Commission, and other Federal agencies, and the emergency action levels that are to be used for determining when and what type of protective measures should be considered within and outside the site boundary to protect health and safety. The emergency action levels shall be based on in-plant conditions and instrumentation in addition to onsite and offsite monitoring. These initial emergency action levels shall be discussed and agreed on by the applicant or licensee and state and local governmental authorities, and approved by the NRC. Thereafter, emergency action levels shall be reviewed with the State and local governmental authorities on an annual basis. A revision to an emergency action level must be approved by the NRC before implementation if:

(1) The licensee is changing from one emergency action level scheme to another emergency action level scheme (*e.g.*, a change from an emergency action level scheme based on NUREG–0654 to a scheme based upon NUMARC/NESP–007 or NEI–99–01);

(2) The licensee is proposing an alternate method for complying with the regulations; or

(3) The emergency action level revision decreases the effectiveness of the emergency plan.

A licensee shall submit each request for NRC approval of the proposed emergency action level change as specified in §50.4. If a licensee makes a change to an EAL that does not require NRC approval, the licensee shall submit, as specified in §50.4, a report of each change made within 30 days after the change is made.

C. Activation of Emergency Organization

The entire spectrum of emergency conditions that involve the alerting or activating of progressively larger segments of the total emergency organization shall be described. The communication steps to be taken to alert or activate emergency personnel under each class of emergency shall be described. Emergency action levels (based not only on onsite and offsite radiation monitoring information but also on readings from a number of sensors that indicate a potential emergency, such as the pressure in containment and the response of the Emergency Core Cooling System) for notification of offsite agencies shall be described. The existence, but not the details, of a message authentication scheme shall be noted for such agencies. The emergency classes defined shall include: (1) notification of unusual events, (2) alert, (3) site area emergency, and (4) general emergency. These classes are further discussed in NUREG–0654; FEMA-REP-1.

D. Notification Procedures

1. Administrative and physical means for notifying local, State, and Federal officials and agencies and agreements reached with these officials and agencies for the prompt notification of the public and for public evacuation or other protective measures, should they become necessary, shall be described. This description shall include identification of the appropriate officials, by title and agency, of the State and local government agencies within the EPZs.[1]

2. Provisions shall be described for yearly dissemination to the public within the plume exposure pathway EPZ of basic emergency planning information, such as the methods and times required for public notification and the protective actions planned if an accident occurs, general information as to the nature and effects of radiation, and a listing of local broadcast stations that will be used for dissemination of information during an emergency. Signs or other measures shall also be used to disseminate to any transient

[1] See footnote 1 to section I.

population within the plume exposure pathway EPZ appropriate information that would be helpful if an accident occurs.

3. A licensee shall have the capability to notify responsible State and local governmental agencies within 15 minutes after declaring an emergency. The licensee shall demonstrate that the State/local officials have the capability to make a public notification decision promptly on being informed by the licensee of an emergency condition. By February 1, 1982, each nuclear power reactor licensee shall demonstrate that administrative and physical means have been established for alerting and providing prompt instructions to the public within the plume exposure pathway EPZ. The four-month period in 10 CFR 50.54(s)(2) for the correction of emergency plan deficiencies shall not apply to the initial installation of this public notification system that is required by February 1, 1982. The four-month period will apply to correction of deficiencies identified during the initial installation and testing of the prompt public notification systems as well as those deficiencies discovered thereafter. The design objective of the prompt public notification system shall be to have the capability to essentially complete the initial notification of the public within the plume exposure pathway EPZ within about 15 minutes. The use of this notification capability will range from immediate notification of the public (within 15 minutes of the time that State and local officials are notified that a situation exists requiring urgent action) to the more likely events where there is substantial time available for the State and local governmental officials to make a judgment whether or not to activate the public notification system. Where there is a decision to activate the notification system, the State and local officials will determine whether to activate the entire notification system simultaneously or in a graduated or staged manner. The responsibility for activating such a public notification system shall remain with the appropriate governmental authorities.

E. Emergency Facilities and Equipment

Adequate provisions shall be made and described for emergency facilities and equipment, including:

1. Equipment at the site for personnel monitoring;

2. Equipment for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials to the environment;

3. Facilities and supplies at the site for decontamination of onsite individuals;

4. Facilities and medical supplies at the site for appropriate emergency first aid treatment;

5. Arrangements for the services of physicians and other medical personnel qualified to handle radiation emergencies on-site;

6. Arrangements for transportation of contaminated injured individuals from the site to specifically identified treatment facilities outside the site boundary;

7. Arrangements for treatment of individuals injured in support of licensed activities on the site at treatment facilities outside the site boundary;

8. A licensee onsite technical support center and a licensee near-site emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency;

9. At least one onsite and one offsite communications system; each system shall have a backup power source.

All communication plans shall have arrangements for emergencies, including titles and alternates for those in charge at both ends of the communication links and the primary and backup means of communication. Where consistent with the function of the governmental agency, these arrangements will include:

a. Provision for communications with contiguous State/local governments within the plume exposure pathway EPZ. Such communications shall be tested monthly.

b. Provision for communications with Federal emergency response organizations. Such communications systems shall be tested annually.

c. Provision for communications among the nuclear power reactor control room, the onsite technical support center, and the near-site emergency operations facility; and among the nuclear facility, the principal State and local emergency operations centers, and the field assessment teams. Such communications systems shall be tested annually.

d. Provisions for communications by the licensee with NRC Headquarters and the appropriate NRC Regional Office Operations Center from the nuclear power reactor control room, the onsite technical support center, and the near-site emergency operations facility. Such communications shall be tested monthly.

F. Training.

1. The program to provide for: (a) The training of employees and exercising, by periodic drills, of radiation emergency plans to ensure that employees of the licensee are familiar with their specific emergency response duties, and (b) The participation in the training and drills by other persons whose assistance may be needed in the event of a radiation emergency shall be described. This shall include a description of specialized initial training and periodic retraining programs to be provided to each of the following categories of emergency personnel:

i. Directors and/or coordinators of the plant emergency organization;

ii. Personnel responsible for accident assessment, including control room shift personnel;

iii Radiological monitoring teams;

iv. Fire control teams (fire brigades);

v. Repair and damage control teams;

vi. First aid and rescue teams;

vii. Medical support personnel;

viii. Licensee's headquarters support personnel;

ix. Security personnel.

In addition, a radiological orientation training program shall be made available to local services personnel; e.g., local emergency services/Civil Defense, local law enforcement personnel, local news media persons.

2. The plan shall describe provisions for the conduct of emergency preparedness exercises as follows: Exercises shall test the adequacy of timing and content of implementing procedures and methods, test emergency equipment and communications networks, test the public notification system, and ensure that emergency organization personnel are familiar with their duties.[3]

a. A full participation[4] exercise which tests as much of the licensee, State and local emergency plans as is reasonably achievable without mandatory public participation shall be conducted for each site at which a power reactor is located. This exercise shall be conducted within two years before the issuance of the first operating license for full power (one authorizing operation above 5% of rated power) of the first reactor and shall include participation by each State and local government within the plume exposure pathway EPZ and each state within the ingestion exposure pathway EPZ. If the full participation exercise is conducted more than one year prior to issuance of an operating licensee for full power, an exercise which tests the licensee's onsite emergency plans shall be conducted within one year before issuance of an operating license for full power. This exercise need not have State or local government participation.

b. Each licensee at each site shall conduct an exercise of its onsite emergency plan every 2 years. The exercise may be included in the full participation biennial exercise required by paragraph 2.c. of this section. In addition, the licensee shall take actions necessary to ensure that adequate emergency response capabilities are maintained during the interval between biennial exercises by conducting drills, including at least one drill involving a combination of some of the principal functional areas of the licensee's onsite emergency response capabilities. The principal functional areas of emergency response include activities such as management and coordination of emergency response, accident assessment, protective action decision-making, and plant system repair and corrective actions. During these drills, activation of all of the licensee's emergency response facilities (Technical Support Center (TSC), Operations Support Center (OSC), and the Emergency Operations Facility (EOF)) would not be necessary, licensees would have the opportunity to consider accident management strategies, supervised instruction would be permitted, operating staff would have the opportunity to resolve problems (success paths) rather than have controllers intervene, and the drills could focus on onsite training objectives.

c. Offsite plans for each site shall be exercised biennially with full participation by each offsite authority having a role under the plan. Where the offsite authority has a role under a radiological response plan for more than one site, it shall fully participate in one exercise every 2 years and shall, at least, partially participate[5] in other offsite plan exercises in this period.

If two different licensees whose licensed facilities are located either on the same site or on adjacent, contiguous sites, and that share most of the elements defining co-located licensees,[6] each licensee shall:

[3] Use of site specific simulators or computers is acceptable for any exercise.

[4] ''Full participation'' when used in conjunction with emergency preparedness exercises for a particular site means appropriate offsite local and State authorities and licensee personnel physically and actively take part in testing their integrated capability to adequately assess and respond to an accident at a commercial nuclear power plant. ''Full participation'' includes testing major observable portions of the onsite and offsite emergency plans and mobilization of state, local and licensee personnel and other resources in sufficient numbers to verify the capability to respond to the accident scenario.

[5] ''Partial participation'' when used in conjunction with emergency preparedness exercises for a particular site means appropriate offsite authorities shall actively take part in the exercise sufficient to test direction and control functions; *i.e.*, (a) protective action decision making related to emergency action levels, and (b) communication capabilities among affected State and local authorities and the licensee.

[6] Co-located licensees are two different licensees whose licensed facilities are located either on the same site or on adjacent, contiguous sites, and that share most of the following emergency planning and siting elements:

Continued

(1) Conduct an exercise biennially of its onsite emergency plan; and

(2) Participate quadrennially in an offsite biennial full or partial participation exercise; and

(3) Conduct emergency preparedness activities and interactions in the years between its participation in the offsite full or partial participation exercise with offsite authorities, to test and maintain interface among the affected state and local authorities and the licensee. Co-located licensees shall also participate in emergency preparedness activities and interaction with offsite authorities for the period between exercises.

d. A State should fully participate in the ingestion pathway portion of exercises at least once every six years. In States with more than one site, the State should rotate this participation from site to site.

e. Licensees shall enable any State or local Government located within the plume exposure pathway EPZ to participate in the licensee's drills when requested by such State or local Government.

f. Remedial exercises will be required if the emergency plan is not satisfactorily tested during the biennial exercise, such that NRC, in consultation with FEMA, cannot find reasonable assurance that adequate protective measures can be taken in the event of a radiological emergency. The extent of State and local participation in remedial exercises must be sufficient to show that appropriate corrective measures have been taken regarding the elements of the plan not properly tested in the previous exercises.

g. All training, including exercises, shall provide for formal critiques in order to identify weak or deficient areas that need correction. Any weaknesses or deficiencies that are identified shall be corrected.

h. The participation of State and local governments in an emergency exercise is not required to the extent that the applicant has identified those governments as refusing to participate further in emergency planning activities, pursuant to 10 CFR 50.47(c)(1). In such cases, an exercise shall be held with the applicant or licensee and such governmental entities as elect to participate in the emergency planning process.

G. Maintaining Emergency Preparedness

Provisions to be employed to ensure that the emergency plan, its implementing procedures, and emergency equipment and supplies are maintained up to date shall be described.

H. Recovery

Criteria to be used to determine when, following an accident, reentry of the facility would be appropriate or when operation could be resumed shall be described.

V. Implementing Procedures

No less than 180 days prior to the scheduled issuance of an operating license for a nuclear power reactor or a license to possess nuclear material the applicant's detailed implementing procedures for its emergency plan shall be submitted to the Commission as specified in § 50.4. Licensees who are authorized to operate a nuclear power facility shall submit any changes to the emergency plan or procedures to the Commission, as specified in § 50.4, within 30 days of such changes.

VI. Emergency Response Data System

1. The Emergency Response Data System (ERDS) is a direct near real-time electronic data link between the licensee's onsite computer system and the NRC Operations Center that provides for the automated transmission of a limited data set of selected parameters. The ERDS supplements the existing voice transmission over the Emergency Notification System (ENS) by providing the NRC Operations Center with timely and accurate updates of a limited set of parameters from the licensee's installed onsite computer system in the event of an emergency. When selected plant data are not available on the licensee's onsite computer system, retrofitting of data points is not required. The licensee shall test the ERDS periodically to verify system availability and operability. The frequency of ERDS testing will be quarterly unless otherwise set by NRC based on demonstrated system performance.

2. Except for Big Rock Point and all nuclear power facilities that are shut down permanently or indefinitely, onsite hardware shall be provided at each unit by the licensee to interface with the NRC receiving system. Software, which will be made available by the NRC, will assemble the data to be transmitted and transmit data from each unit via an output port on the appropriate data system. The hardware and software must have the following characteristics:

a. Data points, if resident in the in-plant computer systems, must be transmitted for four selected types of plant conditions: Reactor core and coolant system conditions; reactor containment conditions; radioactivity release rates; and plant meteorological tower data. A separate data feed is required for each reactor unit. While it is recognized that ERDS is not a safety system, it is conceivable that a licensee's ERDS interface could communicate with a safety system. In

---

a. plume exposure and ingestion emergency planning zones,

b. offsite governmental authorities,

c. offsite emergency response organizations,

d. public notification system, and/or

e. emergency facilities

this case, appropriate isolation devices would be required at these interfaces.[7] The data points, identified in the following parameters will be transmitted:

(i) For pressurized water reactors (PWRs), the selected plant parameters are: (1) Primary coolant system: pressure, temperatures (hot leg, cold leg, and core exit thermocouples), subcooling margin, pressurizer level, reactor coolant charging/makeup flow, reactor vessel level, reactor coolant flow, and reactor power; (2) Secondary coolant system: Steam generator levels and pressures, main feedwater flows, and auxiliary and emergency feedwater flows; (3) Safety injection: High- and low-pressure safety injection flows, safety injection flows (Westinghouse), and borated water storage tank level; (4) Containment: pressure, temperatures, hydrogen concentration, and sump levels; (5) Radiation monitoring system: Reactor coolant radioactivity, containment radiation level, condenser air removal radiation level, effluent radiation monitors, and process radiation monitor levels; and (6) Meteorological data: wind speed, wind direction, and atmospheric stability.

(ii) For boiling water reactors (BWRs), the selected parameters are: (1) Reactor coolant system: Reactor pressure, reactor vessel level, feedwater flow, and reactor power; (2) Safety injection: Reactor core isolation cooling flow, high-pressure coolant injection/high-pressure core spray flow, core spray flow, low-pressure coolant injection flow, and condensate storage tank level; (3) Containment: drywell pressure, drywell temperatures, drywell sump levels, hydrogen and oxygen concentrations, suppression pool temperature, and suppression pool level; (4) Radiation monitoring system: Reactor coolant radioactivity level, primary containment radiation level, condenser off-gas radiation level, effluent radiation monitor, and process radiation levels; and (5) Meteorological data: Wind speed, wind direction, and atmospheric stability.

b. The system must be capable of transmitting all available ERDS parameters at time intervals of not less than 15 seconds or more than 60 seconds. Exceptions to this requirement will be considered on a case by case basis.

c. All link control and data transmission must be established in a format compatible with the NRC receiving system[8] as configured at the time of licensee implementation.

3. Maintaining Emergency Response Data System:

a. Any hardware and software changes that affect the transmitted data points identified in the ERDS Data Point Library[9] (site specific data base residing on the ERDS computer) must be submitted to the NRC within 30 days after the changes are completed.

b. Hardware and software changes, with the exception of data point modifications, that could affect the transmission format and computer communication protocol to the ERDS must be provided to the NRC as soon as practicable and at least 30 days prior to the modification.

c. In the event of a failure of the NRC supplied onsite modem, a replacement unit will be furnished by the NRC for licensee installation.

4. Implementing the Emergency Response Data System Program:

a. Each licensee shall develop and submit an ERDS implementation program plan to the NRC by October 28, 1991. To ensure compatibility with the guidance provided for the ERDS, the ERDS implementation program plan,[10] must include, but not be limited to, information on the licensee's computer system configuration (i.e., hardware and software), interface, and procedures.

b. Licensees must comply with appendix E to part 50, section V.

c. Licensees that have submitted the required information under the voluntary ERDS implementation program will not be required to resubmit this information. The licensee shall meet the implementation schedule of appendix E to part 50, section VI.4d.

d. Each licensee shall complete implementation of the ERDS by February 13, 1993, or before initial escalation to full power, whichever comes later. Licensees with currently operational ERDS interfaces approved under the voluntary ERDS implementation program[11] will not be required to submit another implementation plan and will be considered to have met the requirements for ERDS under appendix E to part 50, section VI.1 and 2 of this part.

[45 FR 55410, Aug. 19, 1980; 46 FR 28839, May 29, 1981, as amended at 46 FR 63032, Dec. 30, 1981; 47 FR 30236, July 13, 1982; 47 FR 57671, Dec. 28, 1982; 49 FR 27736, July 6, 1984; 51 FR 40310, Nov. 6, 1986; 52 FR 16829, May 6, 1987; 52 FR 42086, Nov. 3, 1987; 56 FR 40185, Aug. 13, 1991; 59 FR 14090, Mar. 25, 1994; 61 FR 30132, June 14, 1996]

[7] See 10 CFR 50.55a(h) Protection Systems.

[8] Guidance is provided in NUREG–1394, Revision 1.

[9] See NUREG–1394, Revision 1, appendix C, Data Point Library.

[10] See NUREG–1394, Revision 1, section 3.

[11] See NUREG–1394, Revision 1.

APPENDIX F TO PART 50—POLICY RELATING TO THE SITING OF FUEL REPROCESSING PLANTS AND RELATED WASTE MANAGEMENT FACILITIES

1. Public health and safety considerations relating to licensed fuel reprocessing plants do not require that such facilities be located on land owned and controlled by the Federal Government. Such plants, including the facilities for the temporary storage of high-level radioactive wastes, may be located on privately owned property.

2. A fuel reprocessing plant's inventory of high-level liquid radioactive wastes will be limited to that produced in the prior 5 years. (For the purpose of this statement of policy, ''high-level liquid radioactive wastes'' means those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuels.) High-level liquid radioactive wastes shall be converted to a dry solid as required to comply with this inventory limitation, and placed in a sealed container prior to transfer to a Federal repository in a shipping cask meeting the requirements of 10 CFR part 71. The dry solid shall be chemically, thermally, and radiolytically stable to the extent that the equilibrium pressure in the sealed container will not exceed the safe operating pressure for that container during the period from canning through a minimum of 90 days after receipt (transfer of physical custody) at the Federal repository. All of these high-level radioactive wastes shall be transferred to a Federal repository no later than 10 years following separation of fission products from the irradiated fuel. Upon receipt, the Federal repository will assume permanent custody of these radioactive waste materials although industry will pay the Federal Government a charge which together with interest on unexpended balances will be designed to defray all costs of disposal and perpetual surveillance. The Department of Energy will take title to the radioactive waste material upon transfer to a Federal repository. Before retirement of the reprocessing plant from operational status and before termination of licensing pursuant to §50.82, transfer of all such wastes to a Federal repository shall be completed. Federal repositories, which will be limited in number, will be designated later by the Commission.

3. Disposal of high-level radioactive fission product waste material will not be permitted on any land other than that owned and controlled by the Federal Government.

4. A design objective for fuel reprocessing plants shall be to facilitate decontamination and removal of all significant radioactive wastes at the time the facility is permanently decommissioned. Criteria for the extent of decontamination to be required upon decommissioning and license termination will be developed in consultation with competent groups. Opportunity will be afforded for public comment before such criteria are made effective.

5. Applicants proposing to operate fuel reprocessing plants, in submitting information concerning financial qualifications as required by §50.33(f), shall include information enabling the Commission to determine whether the applicant is financially qualified, among other things, to provide for the removal and disposal of radioactive wastes, during operation and upon decommissioning of the facility, in accordance with the Commission's regulations, including the requirements set out in this appendix.

6. With respect to fuel reprocessing plants already licensed, the licenses will be appropriately conditioned to carry out the purposes of the policy stated above with respect to high-level radioactive fission product wastes generated after installation of new equipment for interim storage of liquid wastes, or after installation of equipment required for solidification without interim liquid storage. In either case, such equipment shall be installed at the earliest practicable date, taking into account the time required for design, procurement and installation thereof. With respect to such plants, the application of the policy stated in this appendix to existing wastes and to wastes generated prior to the installation of such equipment, will be the subject of a further rulemaking proceeding.

[35 FR 17533, Nov. 14, 1970, as amended at 36 FR 5411, Mar. 23, 1971; 42 FR 20139, Apr. 18, 1977; 45 FR 14201, Mar. 5, 1980; 70 FR 3599, Jan. 26, 2005]

APPENDIX G TO PART 50—FRACTURE TOUGHNESS REQUIREMENTS

I. Introduction and scope.
II. Definitions.
III. Fracture toughness tests.
IV. Fracture toughness requirements.

I. INTRODUCTION AND SCOPE

This appendix specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

The ASME Code forms the basis for the requirements of this appendix. ''ASME Code'' means the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. If no section is specified, the reference is to Section III, Division 1, ''Rules for Construction of Nuclear Power Plant Components.'' ''Section XI'' means Section XI, Division 1, ''Rules for Inservice Inspection of Nuclear Power Plant Components.'' If no edition or addenda are specified, the ASME Code edition and addenda and any limitations and modifications thereof, which are specified in §50.55a, are applicable.

The sections, editions and addenda of the ASME Boiler and Pressure Vessel Code specified in §50.55a have been approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the FEDERAL REGISTER. Copies of the ASME Boiler and Pressure Vessel Code may be purchased from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, NY 10017, and are available for inspection at the NRC Library, 11545 Rockville Pike, Two White Flint North, Rockville, MD 20852–2738.

The requirements of this appendix apply to the following materials:

A. Carbon and low-alloy ferritic steel plate, forgings, castings, and pipe with specified minimum yield strengths not over 50,000 psi (345 MPa), and to those with specified minimum yield strengths greater than 50,000 psi (345 MPa) but not over 90,000 psi (621 MPa) if qualified by using methods equivalent to those described in paragraph G–2110 of appendix G of section XI of the latest edition and addenda of the ASME Code incorporated by reference into §50.55a(b)(2).

B. Welds and weld heat-affected zones in the materials specified in paragraph I.A. of this appendix.

C. Materials for bolting and other types of fasteners with specified minimum yield strengths not over 130,000 psi (896 MPa).

NOTE: The adequacy of the fracture toughness of other ferritic materials not covered in this section must be demonstrated to the Director, Office of Nuclear Reactor Regulation, on an individual case basis.

II. DEFINITIONS

A. *Ferritic material* means carbon and low-alloy steels, higher alloy steels including all stainless alloys of the 4xx series, and maraging and precipitation hardening steels with a predominantly body-centered cubic crystal structure.

B. *System hydrostatic tests* means all preoperational system leakage and hydrostatic pressure tests and all system leakage and hydrostatic pressure tests performed during the service life of the pressure boundary in compliance with the ASME Code, Section XI.

C. *Specified minimum yield strength* means the minimum yield strength (in the unirradiated condition) of a material specified in the construction code under which the component is built under §50.55a.

D. RT_{NDT} means the reference temperature of the material, for all conditions.

(i) For the pre-service or unirradiated condition, RT_{NDT} is evaluated according to the procedures in the ASME Code, Paragraph NB–2331.

(ii) For the reactor vessel beltline materials, RT_{NDT} must account for the effects of neutron radiation.

E. ΔRT_{NDT} means the transition temperature shift, or change in RT_{NDT}, due to neutron radiation effects, which is evaluated as the difference in the 30 ft-lb (41 J) index temperatures from the average Charpy curves measured before and after irradiation.

F. *Beltline* or *Beltline region of reactor vessel* means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.

III. FRACTURE TOUGHNESS TESTS

A. To demonstrate compliance with the fracture toughness requirements of section IV of this appendix, ferritic materials must be tested in accordance with the ASME Code and, for the beltline materials, the test requirements of appendix H of this part. For a reactor vessel that was constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition (under §50.55a), the fracture toughness data and data analyses must be supplemented in a manner approved by the Director, Office of Nuclear Reactor Regulation, to demonstrate equivalence with the fracture toughness requirements of this appendix.

B. Test methods for supplemental fracture toughness tests described in paragraph IV.A.1.b of this appendix must be submitted to and approved by the Director, Office of Nuclear Reactor Regulation, prior to testing.

C. All fracture toughness test programs conducted in accordance with paragraphs III.A and III.B must comply with ASME Code requirements for calibration of test equipment, qualification of test personnel, and retention of records of these functions and of the test data.

IV. FRACTURE TOUGHNESS REQUIREMENTS

A. The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials must meet the requirements of the ASME Code, supplemented by the additional requirements set forth below, for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences. Reactor vessels may continue to be operated only for that service period within which the requirements of this section are satisfied. For the reactor vessel beltline materials, including welds, plates and forgings, the values of RT_{NDT} and Charpy upper-shelf energy must account for the effects of neutron radiation, including the results of the surveillance program of appendix H of this part. The effects of neutron radiation must consider the radiation conditions (i.e., the fluence) at the deepest point on the crack front of the flaw assumed in the analysis.

1. Reactor Vessel Charpy Upper-Shelf Energy Requirements

a. Reactor vessel beltline materials must have Charpy upper-shelf energy,[1] in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 75 ft-lb (102 J) initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68 J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. This analysis must use the latest edition and addenda of the ASME Code incorporated by reference into §50.55a(b)(2) at the time the analysis is submitted.

b. Additional evidence of the fracture toughness of the beltline materials after exposure to neutron irradiation may be obtained from results of supplemental fracture toughness tests for use in the analysis specified in section IV.A.1.a.

c. The analysis for satisfying the requirements of section IV.A.1 of this appendix must be submitted, as specified in §50.4, for review and approval on an individual case basis at least three years prior to the date when the predicted Charpy upper-shelf energy will no longer satisfy the requirements of section IV.A.1 of this appendix, or on a schedule approved by the Director, Office of Nuclear Reactor Regulation.

2. Pressure-Temperature Limits and Minimum Temperature Requirements

a. Pressure-temperature limits and minimum temperature requirements for the reactor vessel are given in table 3, and are defined by the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational occurrences), the vessel pressure, whether or not fuel is in the vessel, and whether the core is critical. In table 3, the vessel pressure is defined as a percentage of the preservice system hydrostatic test pressure. The appropriate requirements on both the pressure-temperature limits and the minimum permissible temperature must be met for all conditions.

b. The pressure-temperature limits identified as ''ASME Appendix G limits'' in table 3 require that the limits must be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of Appendix G of Section XI of the ASME Code.

c. The minimum temperature requirements given in table 3 pertain to the controlling material, which is either the material in the closure flange or the material in the beltline region with the highest reference temperature. As specified in table 3, the minimum temperature requirements and the controlling material depend on the operating condition (i.e., hydrostatic pressure and leak tests, or normal operation including anticipated operational occurrences), the vessel pressure, whether fuel is in the vessel, and whether the core is critical. The metal temperature of the controlling material, in the region of the controlling material which has the least favorable combination of stress and temperature, must exceed the appropriate minimum temperature requirement for the condition and pressure of the vessel specified in table 1.

d. Pressure tests and leak tests of the reactor vessel that are required by Section XI of the ASME Code must be completed before the core is critical.

B. If the procedures of section IV.A. of this appendix do not indicate the existence of an equivalent safety margin, the reactor vessel beltline may be given a thermal annealing treatment to recover the fracture toughness of the material, subject to the requirements of §50.66. The reactor vessel may continue to be operated only for that service period within which the predicted fracture toughness of the beltline region materials satisfies the requirements of section IV.A. of this appendix using the values of RT_{NDT} and Charpy upper-shelf energy that include the effects of annealing and subsequent irradiation.

[1] Defined in ASTM E 185–79 and –82 which are incorporated by reference in appendix H to part 50.

TABLE 1—PRESSURE AND TEMPERATURE REQUIREMENTS FOR THE REACTOR PRESSURE VESSEL

Operating condition	Vessel pressure [1]	Requirements for pressure-temperature limits	Minimum temperature requirements
1. Hydrostatic pressure and leak tests (core is not critical):			
1.a Fuel in the vessel	≤20%	ASME Appendix G Limits	(2)
1.b Fuel in the vessel	>20%	ASME Appendix G Limits	(2) +90 °F (6)
1.c No fuel in the vessel (Preservice Hydrotest Only).	ALL	(Not Applicable)	(3) +60 °F
2. Normal operation (incl. heat-up and cool-down), including anticipated operational occurrences:			
2.a Core not critical	≤20%	ASME Appendix G Limits	(2)
2.b Core not critical	>20%	ASME Appendix G Limits	(2) +120 °F (6)
2.c Core critical	≤20%	ASME Appendix G Limits + 40 °F	Larger of [(4)] or [(2) + 40 °F]
2.d Core critical	>20%	ASME Appendix G Limits + 40 °F	Larger of [(4)] or [(2) + 160 °F]
2.e Core critical for BWR (5)	≤20%	ASME Appendix G Limits + 40 °F	(2) + 60 °F

[1] Percent of the preservice system hydrostatic test pressure.
[2] The highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload.
[3] The highest reference temperature of the vessel.
[4] The minimum permissible temperature for the inservice system hydrostatic pressure test.
[5] For boiling water reactors (BWR) with water level within the normal range for power operation.
[6] Lower temperatures are permissible if they can be justified by showing that the margins of safety of the controlling region are equivalent to those required for the beltline when it is controlling.

60 FR 65474, Dec. 19, 1995]

APPENDIX H TO PART 50—REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM REQUIREMENTS

I. Introduction
II. Definitions
III. Surveillance Program Criteria
IV. Report of Test Results

I. INTRODUCTION

The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the reactor vessel. These data will be used as described in section IV of appendix G to part 50.

ASTM E 185–73, ''Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels''; ASTM E 185–79, ''Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels''; and ASTM E 185–82, ''Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels''; which are referenced in the following paragraphs, have been approved for incorporation by reference by the Director of the Federal Register. Copies of ASTM E 185–73, –79, and –82, may be purchased from the American Society for Testing and Materials, 1916 Race Street, Philadelphia, PA 19103 and are available for inspection at the NRC Library, 11545 Rockville Pike, Two White Flint North, Rockville, MD 20852–2738.

II. DEFINITIONS

All terms used in this appendix have the same meaning as in appendix G.

III. SURVEILLANCE PROGRAM CRITERIA

A. No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence at the end of the design life of the vessel will not exceed 10^{17} n/cm^2 (E > 1 MeV).

B. Reactor vessels that do not meet the conditions of paragraph III.A of this appendix must have their beltline materials monitored by a surveillance program complying with ASTM E 185, as modified by this appendix.

1. The design of the surveillance program and the withdrawal schedule must meet the requirements of the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased. Later editions of ASTM E 185 may be used, but including only those editions through 1982. For each capsule withdrawal, the test procedures and reporting requirements must meet the requirements of

ASTM E 185–82 to the extent practicable for the configuration of the specimens in the capsule.

2. Surveillance specimen capsules must be located near the inside vessel wall in the beltline region so that the specimen irradiation history duplicates, to the extent practicable within the physical constraints of the system, the neutron spectrum, temperature history, and maximum neutron fluence experienced by the reactor vessel inner surface. If the capsule holders are attached to the vessel wall or to the vessel cladding, construction and inservice inspection of the attachments and attachment welds must be done according to the requirements for permanent structural attachments to reactor vessels given in Sections III and XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). The design and location of the capsule holders must permit insertion of replacement capsules. Accelerated irradiation capsules may be used in addition to the required number of surveillance capsules.

3. A proposed withdrawal schedule must be submitted with a technical justification as specified in §50.4. The proposed schedule must be approved prior to implementation.

C. Requirements for an Integrated Surveillance Program.

1. In an integrated surveillance program, the representative materials chosen for surveillance for a reactor are irradiated in one or more other reactors that have similar design and operating features. Integrated surveillance programs must be approved by the Director, Office of Nuclear Reactor Regulation, on a case-by-case basis. Criteria for approval include the following:

a. The reactor in which the materials will be irradiated and the reactor for which the materials are being irradiated must have sufficiently similar design and operating features to permit accurate comparisons of the predicted amount of radiation damage.

b. Each reactor must have an adequate dosimetry program.

c. There must be adequate arrangement for data sharing between plants.

d. There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

e. There must be substantial advantages to be gained, such as reduced power outages or reduced personnel exposure to radiation, as a direct result of not requiring surveillance capsules in all reactors in the set.

2. No reduction in the requirements for number of materials to be irradiated, specimen types, or number of specimens per reactor is permitted.

3. After (the effective date of this section), no reduction in the amount of testing is permitted unless previously authorized by the Director, Office of Nuclear Reactor Regulation.

IV. Report of Test Results

A. Each capsule withdrawal and the test results must be the subject of a summary technical report to be submitted, as specified in §50.4, within one year of the date of capsule withdrawal, unless an extension is granted by the Director, Office of Nuclear Reactor Regulation.

B. The report must include the data required by ASTM E 185, as specified in paragraph III.B.1 of this appendix, and the results of all fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions.

C. If a change in the Technical Specifications is required, either in the pressure-temperature limits or in the operating procedures required to meet the limits, the expected date for submittal of the revised Technical Specifications must be provided with the report.

[60 FR 65476, Dec. 19, 1995, as amended at 68 FR 75390, Dec. 31, 2003]

Appendix I to Part 50—Numerical Guides for Design Objectives and Limiting Conditions for Operation To Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents

SECTION I. *Introduction.* Section 50.34a provides that an application for a permit to construct a nuclear power reactor shall include a description of the preliminary design of equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents produced during normal conditions, including expected occurrences. In the case of an application filed on or after January 2, 1971, the application must also identify the design objectives, and the means to be employed, for keeping levels of radioactive material in effluents to unrestricted areas as low as practicable.

Section 50.36a contains provisions designed to assure that releases of radioactive material from nuclear power reactors to unrestricted areas during normal conditions, including expected occurrences, are kept as low as practicable.

SEC. II. *Guides on design objectives for light-water-cooled nuclear power reactors licensed under 10 CFR part 50.* The guides on design objectives set forth in this section may be used by an applicant for a permit to construct a light-water-cooled nuclear power reactor as guidance in meeting the requirements of §50.34a(a). The applicant shall provide reasonable assurance that the following design objectives will be met.

A. The calculated annual total quantity of all radioactive material above background[1] to be released from each light-water-cooled nuclear power reactor to unrestricted areas will not result in an estimated annual dose or dose commitment from liquid effluents for any individual in an unrestricted area from all pathways of exposure in excess of 3 millirems to the total body or 10 millirems to any organ.

B.1. The calculated annual total quantity of all radioactive material above background to be released from each light-water-cooled nuclear power reactor to the atmosphere will not result in an estimated annual air dose from gaseous effluents at any location near ground level which could be occupied by individuals in unrestricted areas in excess of 10 millirads for gamma radiation or 20 millirads for beta radiation.

2. Notwithstanding the guidance of paragraph B.1:

(a) The Commission may specify, as guidance on design objectives, a lower quantity of radioactive material above background to be released to the atmosphere if it appears that the use of the design objectives in paragraph B.1 is likely to result in an estimated annual external dose from gaseous effluents to any individual in an unrestricted area in excess of 5 millirems to the total body; and

(b) Design objectives based upon a higher quantity of radioactive material above background to be released to the atmosphere than the quantity specified in paragraph B.1 will be deemed to meet the requirements for keeping levels of radioactive material in gaseous effluents as low as is reasonably achievable if the applicant provides reasonable assurance that the proposed higher quantity will not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas in excess of 5 millirems to the total body or 15 millirems to the skin.

C. The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form above background to be released from each light-water-cooled nuclear power reactor in effluents to the atmosphere will not result in an estimated annual dose or dose commitment from such radioactive iodine and radioactive material in particulate form for any individual in an unrestricted area from all pathways of exposure in excess of 15 millirems to any organ.

D. In addition to the provisions of paragraphs A, B, and C above, the applicant shall include in the radwaste system all items of reasonably demonstrated technology that, when added to the system sequentially and in order of diminishing cost-benefit return, can for a favorable cost-benefit ratio effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor. As an interim measure and until establishment and adoption of better values (or other appropriate criteria), the values $1000 per total body man-rem and $1000 per man-thyroid-rem (or such lesser values as may be demonstrated to be suitable in a particular case) shall be used in this cost-benefit analysis. The requirements of this paragraph D need not be complied with by persons who have filed applications for construction permits which were docketed on or after January 2, 1971, and prior to June 4, 1976, if the radwaste systems and equipment described in the preliminary or final safety analysis report and amendments thereto satisfy the Guides on Design Objectives for Light-Water-Cooled Nuclear Power Reactors proposed in the Concluding Statement of Position of the Regulatory Staff in Docket-RM-50-2 dated February 20, 1974, pp. 25–30, reproduced in the annex to this appendix I.

SEC. III. *Implementation*. A.1. Conformity with the guides on design objectives of Section II shall be demonstrated by calculational procedures based upon models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated, all uncertainties being considered together. Account shall be taken of the cumulative effect of all sources and pathways within the plant contributing to the particular type of effluent being considered. For determination of design objectives in accordance with the guides of Section II, the estimations of exposure shall be made with respect to such potential land and water usage and food pathways as could actually exist during the term of plant operation: *Provided*, That, if the requirements of paragraph B of Section III are fulfilled, the applicant shall be deemed to have complied with the requirements of paragraph C of Section II with respect to radioactive iodine if estimations of exposure are made on the basis of such food pathways and individual receptors as actually exist at the time the plant is licensed.

2. The characteristics attributed to a hypothetical receptor for the purpose of estimating internal dose commitment shall take

[1] Here and elsewhere in this appendix background means radioactive materials in the environment and in the effluents from light-water-cooled power reactors not generated in, or attributable to, the reactors of which specific account is required in determining design objectives.

into account reasonable deviations of individual habits from the average. The applicant may take account of any real phenomenon or factors actually affecting the estimate of radiation exposure, including the characteristics of the plant, modes of discharge of radioactive materials, physical processes tending to attenuate the quantity of radioactive material to which an individual would be exposed, and the effects of averaging exposures over times during which determining factors may fluctuate.

B. If the applicant determines design objectives with respect to radioactive iodine on the basis of existing conditions and if potential changes in land and water usage and food pathways could result in exposures in excess of the guideline values of paragraph C of Section II, the applicant shall provide reasonable assurance that a monitoring and surveillance program will be performed to determine:

1. The quantities of radioactive iodine actually released to the atmosphere and deposited relative to those estimated in the determination of design objectives;

2. Whether changes in land and water usage and food pathways which would result in individual exposures greater than originally estimated have occurred; and

3. The content of radioactive iodine and foods involved in the changes, if and when they occur.

SEC. IV. *Guides on technical specifications for limiting conditions for operation for light-water-cooled nuclear power reactors licensed under 10 CFR part 50.* The guides on limiting conditions for operation for light-water-cooled nuclear power reactors set forth below may be used by an applicant for a license to operate a light-water-cooled nuclear power reactor or a licensee who has submitted a certification of permanent cessation of operations under §50.82(a)(1) as guidance in developing technical specifications under §50.36a(a) to keep levels of radioactive materials in effluents to unrestricted areas as low as is reasonably achievable.

Section 50.36a(b) provides that licensees shall be guided by certain considerations in establishing and implementing operating procedures specified in technical specifications that take into account the need for operating flexibility and at the same time assure that the licensee will exert his best effort to keep levels of radioactive material in effluents as low as is reasonably achievable. The guidance set forth below provides additional and more specific guidance to licensees in this respect.

Through the use of the guides set forth in this section it is expected that the annual release of radioactive material in effluents from light-water-cooled nuclear power reactors can generally be maintained within the levels set forth as numerical guides for design objectives in Section II.

At the same time, the licensee is permitted the flexibility of operations, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power even under unusual conditions which may temporarily result in releases higher than numerical guides for design objectives but still within levels that assure that the average population exposure is equivalent to small fractions of doses from natural background radiation. It is expected that in using this operational flexibility under unusual conditions, the licensee will exert his best efforts to keep levels of radioactive material in effluents within the numerical guides for design objectives.

A. If the quantity of radioactive material actually released in effluents to unrestricted areas from a light-water-cooled nuclear power reactor during any calendar quarter is such that the resulting radiation exposure, calculated on the same basis as the respective design objective exposure, would exceed one-half the design objective annual exposure derived pursuant to Sections II and III, the licensee shall:[2]

1. Make an investigation to identify the causes for such release rates;

2. Define and initiate a program of corrective action; and

3. Report these actions as specified in §50.4, within 30 days from the end of the quarter during which the release occurred.

B. The licensee shall establish an appropriate surveillance and monitoring program to:

1. Provide data on quantities of radioactive material released in liquid and gaseous effluents to assure that the provisions of paragraph A of this section are met;

2. Provide data on measurable levels of radiation and radioactive materials in the environment to evaluate the relationship between quantities of radioactive material released in effluents and resultant radiation doses to individuals from principal pathways of exposure; and

3. Identify changes in the use of unrestricted areas (e.g., for agricultural purposes) to permit modifications in monitoring programs for evaluating doses to individuals from principal pathways of exposure.

[2] Section 50.36a(a)(2) requires the licensee to submit certain reports to the Commission with regard to the quantities of the principal radionuclides released to unrestricted areas. It also provides that, on the basis of such reports and any additional information the Commission may obtain from the licensee and others, the Commission may from time to time require the license to take such action as the Commission deems appropriate.

C. If the data developed in the surveillance and monitoring program described in paragraph B of Section III or from other monitoring programs show that the relationship between the quantities of radioactive material released in liquid and gaseous effluents and the dose to individuals in unrestricted areas is significantly different from that assumed in the calculations used to determine design objectives pursuant to Sections II and III, the Commission may modify the quantities in the technical specifications defining the limiting conditions in a license to operate a light-water-cooled nuclear power reactor or a license whose holder has submitted a certification of permanent cessation of operations under §50.82(a)(1).

SEC. V. *Effective dates.* A. The guides for limiting conditions for operation set forth in this appendix shall be applicable in any case in which an application was filed on or after January 2, 1971, for a permit to construct a light-water-cooled nuclear power reactor.

B. For each light-water-cooled nuclear power reactor constructed pursuant to a permit for which application was filed prior to January 2, 1971, the holder of the permit or a license, authorizing operation of the reactor shall, within a period of twelve months from June 4, 1975, file with the Commission:

1. Such information as is necessary to evaluate the means employed for keeping levels of radioactivity in effluents to unrestricted areas as low as is reasonably achievable, including all such information as is required by §50.34a (b) and (c) not already contained in his application; and

2. Plans and proposed technical specifications developed for the purpose of keeping releases of radioactive materials to unrestricted areas during normal reactor operations, including expected operational occurrences, as low as is reasonably achievable.

CONCLUDING STATEMENT OF POSITION OF THE REGULATORY STAFF (DOCKET-RM-50-2)

GUIDES ON DESIGN OBJECTIVES FOR LIGHT-WATER-COOLED NUCLEAR POWER REACTORS

A. For radioactive material above background[1] in liquid effluents to be released to unrestricted areas:

1. The calculated annual total quantity of all radioactive material from all light-water-cooled nuclear power reactors at a site should not result in an annual dose or dose commitment to the total body or to any organ of an individual in an unrestricted area from all pathways of exposure in excess of 5 millirems; and

2. The calculated annual total quantity of radioactive material, except tritium and dissolved gases, should not exceed 5 curies for each light-water-cooled reactor at a site.

3. Notwithstanding the guidance in paragraph A.2, for a particular site, if an applicant for a permit to construct a light-water-cooled nuclear power reactor has proposed baseline in-plant control measures[2] to reduce the possible sources of radioactive material in liquid effluent releases and the calculated quantity exceeds the quantity set forth in paragraph A.2, the requirements for design objectives for radioactive material in liquid effluents may be deemed to have been met provided:

a. The applicant submits, as specified in §50.4, an evaluation of the potential for effects from long-term buildup on the environment in the vicinity of the site of radioactive material, with a radioactive half-life greater than one year, to be released; and

b. The provisions of paragraph A.1 are met.

B. For radioactive material above background in gaseous effluents the annual total quantity of radioactive material to be released to the atmosphere by all light-water-cooled nuclear power reactors at a site:

1. The calculated annual air dose due to gamma radiation at any location near ground level which could be occupied by individuals at or beyond the boundary of the site should not exceed 10 millirads; and

2. The calculated annual air dose due to beta radiation at any location near ground level which could be occupied by individuals at or beyond the boundary of the site should not exceed 20 millirads.

3. Notwithstanding the guidance in paragraphs B.1 and B.2, for a particular site:

a. The Commission may specify, as guidance on design objectives, a lower quantity of radioactive material above background in gaseous effluents to be released to the atmosphere if it appears that the use of the design objectives described in paragraphs B.1 and B.2 is likely to result in an annual dose to an individual in an unrestricted area in excess of 5 millirems to the total body or 15 millirems to the skin; or

b. Design objectives based on a higher quantity of radioactive material above background in gaseous effluents to be released to

[1] "Background," means the quantity of radioactive material in the effluent from light-water-cooled nuclear power reactors at a site that did not originate in the reactors.

[2] Such measures may include treatment of clear liquid waste streams (normally tritiated, nonaerated, low conductivity equipment drains and pump seal leakoff), dirty liquid waste streams (normally nontritiated, aerated, high conductivity building sumps, floor and sample station drains), steam generator blowdown streams, chemical waste streams, low purity and high purity liquid streams (resin regenerate and laboratory wastes), as appropriate for the type of reactor.

the atmosphere than the quantity specified in paragraphs B.1 and B.2 may be deemed to meet the requirements for keeping levels of radioactive material in gaseous effluents as low as practicable if the applicant provides reasonable assurance that the proposed higher quantity will not result in annual doses to an individual in an unrestricted area in excess of 5 millirems to the total body or 15 millirems to the skin.

C. For radioactive iodine and radioactive material in particulate form above background released to the atmosphere:

1. The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form from all light-water-cooled nuclear power reactors at a site should not result in an annual dose or dose commitment to any organ of an individual in an unrestricted area from all pathways of exposure in excess of 15 millirems. In determining the dose or dose commitment the portion thereof due to intake of radioactive material via the food pathways may be evaluated at the locations where the food pathways actually exist; and

2. The calculated annual total quantity of iodine-131 in gaseous effluents should not exceed 1 curie for each light-water-cooled nuclear power reactor at a site.

3. Notwithstanding the guidance in paragraphs C.1 and C.2 for a particular site, if an applicant for a permit to construct a light-water-cooled nuclear power reactor has proposed baseline in-plant control measures[3] to reduce the possible sources of radioactive iodine releases, and the calculated annual quantities taking into account such control measures exceed the design objective quantities set forth in paragraphs C.1 and C.2, the requirements for design objectives for radioactive iodine and radioactive material in particulate form in gaseous effluents may be deemed to have been met provided the calculated annual total quantity of all radioactive iodine and radioactive material in particulate form that may be released in gaseous effluents does not exceed four times the quantity calculated pursuant to paragraph C.1.

[40 FR 19442, May 5, 1975, as amended at 40 FR 40818, Sept. 4, 1975; 40 FR 58847, Dec. 19, 1975; 41 FR 16447, Apr. 19, 1976; 42 FR 20139, Apr. 18, 1977; 51 FR 40311, Nov. 6, 1986; 61 FR 39303, July 29, 1996]

[3] Such in-plant control measures may include treatment of steam generator blowdown tank exhaust, clean steam supplies for turbine gland seals, condenser vacuum systems, containment purging exhaust and ventilation exhaust systems and special design features to reduce contaminated steam and liquid leakage from valves and other sources such as sumps and tanks, as appropriate for the type of reactor.

APPENDIX J TO PART 50—PRIMARY REACTOR CONTAINMENT LEAKAGE TESTING FOR WATER-COOLED POWER REACTORS

This appendix includes two options, A and B, either of which can be chosen for meeting the requirements of this appendix.

OPTION A—PRESCRIPTIVE REQUIREMENTS

Table of Contents

I. INTRODUCTION

One of the conditions of all operating licenses for water-cooled power reactors as specified in §50.54(o) is that primary reactor containments shall meet the containment leakage test requirements set forth in this appendix. These test requirements provide for preoperational and periodic verification by tests of the leak-tight integrity of the primary reactor containment, and systems and components which penetrate containment of water-cooled power reactors, and establish the acceptance criteria for such tests. The purposes of the tests are to assure that (a) leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values as specified in the technical specifications or associated bases and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment. These test requirements may also be used for guidance in establishing appropriate containment leakage test requirements in technical specifications or associated bases for other types of nuclear power reactors.

II. EXPLANATION OF TERMS

A. ''Primary reactor containment'' means the structure or vessel that encloses the components of the reactor coolant pressure boundary, as defined in §50.2(v), and serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.

B. ''Containment isolation valve'' means any valve which is relied upon to perform a containment isolation function.

C. ''Reactor containment leakage test program'' includes the performance of Type A, Type B, and Type C tests, described in II.F, II.G, and II.H, respectively.

D. ''Leakage rate'' for test purposes is that leakage which occurs in a unit of time, stated as a percentage of weight of the original content of containment air at the leakage rate test pressure that escapes to the outside atmosphere during a 24-hour test period.

E. ''Overall integrated leakage rate'' means that leakage rate which obtains from a summation of leakage through all potential leakage paths including containment welds, valves, fittings, and components which penetrate containment.

F. ''Type A Tests'' means tests intended to measure the primary reactor containment overall integrated leakage rate (1) after the containment has been completed and is ready for operation, and (2) at periodic intervals thereafter.

G. ''Type B Tests'' means tests intended to detect local leaks and to measure leakage across each pressure-containing or leakage-limiting boundary for the following primary reactor containment penetrations:

1. Containment penetrations whose design incorporates resilient seals, gaskets, or sealant componds, piping penetrations fitted with expansion bellows, and electrical penetrations fitted with flexible metal seal assemblies.

2. Air lock door seals, including door operating mechanism penetrations which are part of the containment pressure boundary.

3. Doors with resilient seals or gaskets except for seal-welded doors.

4. Components other than those listed in II.G.1, II.G.2, or II.G.3 which must meet the acceptance criteria in III.B.3.

H. ''Type C Tests'' means tests intended to measure containment isolation valve leakage rates. The containment isolation valves included are those that:

1. Provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation, such as purge and ventilation, vacuum relief, and instrument valves;

2. Are required to close automatically upon receipt of a containment isolation signal in response to controls intended to effect containment isolation;

3. Are required to operate intermittently under postaccident conditions; and

4. Are in main steam and feedwater piping and other systems which penetrate containment of direct-cycle boiling water power reactors.

I. Pa (p.s.i.g.) means the calculated peak containment internal pressure related to the design basis accident and specified either in the technical specification or associated bases.

J. Pt (p.s.i.g.) means the containment vessel reduced test pressure selected to measure the integrated leakage rate during periodic Type A tests.

K. La (percent/24 hours) means the maximum allowable leakage rate at pressure Pa as specified for preoperational tests in the technical specifications or associated bases, and as specified for periodic tests in the operating license.

L. Ld (percent/24 hours) means the design leakage rate at pressure, Pa, as specified in the technical specifications or associated bases.

M. Lt (percent/24 hours) means the maximum allowable leakage rate at pressure Pt derived from the preoperational test data as specified in III.A.4.(a)(iii).

N. Lam, Ltm (percent/24 hours) means the total measured containment leakage rates at pressure Pa and Pt, respectively, obtained from testing the containment with components and systems in the state as close as practical to that which would exist under design basis accident conditions (e.g., vented, drained, flooded or pressurized).

O. ''Acceptance criteria'' means the standard against which test results are to be compared for establishing the functional acceptability of the containment as a leakage limiting boundary.

III. LEAKAGE TESTING REQUIREMENTS

A program consisting of a schedule for conducting Type A, B, and C tests shall be developed for leak testing the primary reactor containment and related systems and components penetrating primary containment pressure boundary.

Upon completion of construction of the primary reactor containment, including installation of all portions of mechanical, fluid, electrical, and instrumentation systems penetrating the primary reactor containment pressure boundary, and prior to any reactor operating period, preoperational and periodic leakage rate tests, as applicable, shall be conducted in accordance with the following:

A. *Type A test*—1. *Pretest requirements*. (a) Containment inspection in accordance with V. A. shall be performed as a prerequisite to the performance of Type A tests. During the period between the initiation of the containment inspection and the performance of the Type A test, no repairs or adjustments shall be made so that the containment can be tested in as close to the ''as is'' condition as practical. During the period between the completion of one Type A test and the initiation of the containment inspection for the subsequent Type A test, repairs or adjustments shall be made to components whose leakage exceeds that specified in the technical specification as soon as practical after identification. If during a Type A test, including the supplemental test specified in III.A.3.(b), potentially excessive leakage paths are identified which will interfere with satisfactory completion of the test, or which result in the Type A test not meeting the acceptance criteria III.A.4.(b) or III.A.5.(b), the Type A test shall be terminated and the leakage through such paths shall be measured using local leakage testing methods. Repairs and/or adjustments to equipment shall be made and Type A test performed. The corrective action taken and the change in leakage rate determined from the tests and overall integrated leakage determined from local leak and Type A tests shall be included in the summary report required by V.B.

(b) Closure of containment isolation valves for the Type A test shall be accomplished by normal operation and without any preliminary exercising or adjustments (e.g., no tightening of valve after closure by valve motor). Repairs of maloperating or leaking valves shall be made as necessary. Information on any valve closure malfunction or valve leakage that require corrective action before the test, shall be included in the summary report required by V.B.

(c) The containment test conditions shall stabilize for a period of about 4 hours prior to the start of a leakage rate test.

(d) Those portions of the fluid systems that are part of the reactor coolant pressure boundary and are open directly to the containment atmosphere under post-accident conditions and become an extension of the boundary of the containment shall be opened or vented to the containment atmosphere prior to and during the test. Portions of closed systems inside containment that penetrate containment and rupture as a result of a loss of coolant accident shall be vented to the containment atmosphere. All vented systems shall be drained of water or other fluids to the extent necessary to assure exposure of the system containment isolation valves to containment air test pressure and to assure they will be subjected to the post accident differential pressure. Systems that are required to maintain the plant in a safe condition during the test shall be operable in their normal mode, and need not be vented. Systems that are normally filled with water and operating under post-accident conditions, such as the containment heat removal system, need not be vented. However, the containment isolation valves in the systems defined in III.A.1.(d) shall be tested in accordance with III.C. The measured leakage rate from these tests shall be included in the summary report required by V.B.

2. *Conduct of tests.* Preoperational leakage rate tests at either reduced or at peak pressure, shall be conducted at the intervals specified in III.D.

3. *Test Methods.* (a) All Type A tests shall be conducted in accordance with the provisions of the American National Standards N45.4–1972, ''Leakage Rate Testing of Containment Structures for Nuclear Reactors,'' March 16, 1972. In addition to the Total time and Point-to-Point methods described in that standard, the Mass Point Method, when used with a test duration of at least 24 hours, is an acceptable method to use to calculate leakage rates. A typical description of the Mass Point method can be found in the American National Standard ANSI/ANS 56.8–1987, ''Containment System Leakage Testing Requirements,'' January 20, 1987. Incorporation of ANSI N45.4–1972 by reference was approved by the Director of the Federal Register. Copies of this standard, as well as ANSI/ANS–56.8–1987, ''Containment System Leakage Testing Requirements'' (dated January 20, 1987) may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, IL 60525. A copy of each of these standards is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852–2738.

(b) The accuracy of any Type A test shall be verified by a supplemental test. An acceptable method is described in Appendix C of ANSI N45.4–1972. The supplemental test method selected shall be conducted for sufficient duration to establish accurately the change in leakage rate between the Type A and supplemental test. Results from this supplemental test are acceptable provided the difference between the supplemental test data and the Type A test data is within 0.25 La (or 0.25 Lt). If results are not within 0.25 La (or 0.25 Lt), the reason shall be determined, corrective action taken, and a successful supplemental test performed.

(c) Test leakage rates shall be calculated using absolute values corrected for instrument error.

4. *Preoperational leakage rate tests.* (a) *Test pressure*—(1) *Reduced pressure tests.* (i) An initial test shall be performed at a pressure Pt, not less than 0.50 Pa to measure a leakage rate Ltm.

(ii) A second test shall be performed at pressure Pa to measure a leakage rate Lam.

(iii) The leakage characteristics yielded by measurements Ltm and Lam shall establish the maximum allowable test leakage rate Lt of not more than La (Ltm/Lam). In the event Ltm/Lam is greater than 0.7, Lt shall be specified as equal to La (Pt/Pa).[1]

(2) *Peak pressure tests.* A test shall be performed at pressure Pa to measure the leakage rate Lam.

[1] Such inservice inspections are required by § 50.55a.

(b) *Acceptance criteria*—(1) *Reduced pressure tests*. The leakage rate Ltm shall be less than 0.75 Lt.

(2) *Peak pressure tests*. The leakage rate Lam shall be less than 0.75 La and not greater than Ld.

5. *Periodic leakage rate tests*—(a) *Test pressure*. (1) Reduced pressure tests shall be conducted at Pt;

(2) Peak pressure tests shall be conducted at Pa.

(b) *Acceptance criteria*—(1) *Reduced pressure tests*. The leakage rate Ltm shall be less than 0.75 Lt. If local leakage measurements are taken to effect repairs in order to meet the acceptance criteria, these measurements shall be taken at a test pressure Pt.

(2) *Peak pressure tests*. The leakage rate Lam shall be less than 0.75 La. If local leakage measurements are taken to effect repairs in order to meet the acceptance criteria, these measurements shall be taken at a test pressure Pa.

6. *Additional requirements*. (a) If any periodic Type A test fails to meet the applicable acceptance criteria in III.A.5.(b), the test schedule applicable to subsequent Type A tests will be reviewed and approved by the Commission.

(b) If two consecutive periodic Type A tests fail to meet the applicable acceptance criteria in III.A.5(b), notwithstanding the periodic retest schedule of III.D., a Type A test shall be performed at each plant shutdown for refueling or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet the acceptance criteria in III.A.5(b), after which time the retest schedule specified in III.D. may be resumed.

B. *Type B tests*—1. *Test methods*. Acceptable means of performing preoperation and periodic Type B tests include:

(a) Examination by halide leak-detection method (or by other equivalent test methods such as mass spectrometer) of a test chamber, pressurized with air, nitrogen, or pneumatic fluid specified in the technical specifications or associated bases and constructed as part of individual containment penetrations.

(b) Measurement of the rate of pressure loss of the test chamber of the containment penetration pressurized with air, nitrogen, or pneumatic fluid specified in the technical specifications or associated bases.

(c) Leakage surveillance by means of a permanently installed system with provisions for continuous or intermittent pressurization of individual or groups of containment penetrations and measurement of rate of pressure loss of air, nitrogen, or pneumatic fluid specified in the technical specification or associated bases through the leak paths.

2. *Test pressure*. All preoperational and periodic Type B tests shall be performed by local pneumatic pressurization of the containment penetrations, either individually or in groups, at a pressure not less than Pa.

3. *Acceptance criteria*. (See also Type C tests.) (a) The combined leakage rate of all penetrations and valves subject to Type B and C tests shall be less than 0.60 La, with the exception of the valves specified in III.C.3.

(b) Leakage measurements obtained through component leakage surveillance systems (e.g., continuous pressurization of individual containment components) that maintains a pressure not less than Pa at individual test chambers of containment penetrations during normal reactor operation, are acceptable in lieu of Type B tests.

C. *Type C tests*—1. *Test method*. Type C tests shall be performed by local pressurization. The pressure shall be applied in the same direction as that when the value would be required to perform its safety function, unless it can be determined that the results from the tests for a pressure applied in a different direction will provide equivalent or more conservative results. The test methods in III.B.1 may be substituted where appropriate. Each valve to be tested shall be closed by normal operation and without any preliminary exercising or adjustments (e.g., no tightening of valve after closure by valve motor).

2. *Test pressure*. (a) Valves, unless pressurized with fluid (e.g., water, nitrogen) from a seal system, shall be pressurized with air or nitrogen at a pressure of Pa.

(b) Valves, which are sealed with fluid from a seal system shall be pressurized with that fluid to a pressure not less than 1.10 Pa.

3. *Acceptance criterion*. The combined leakage rate for all penetrations and valves subject to Type B and C tests shall be less than 0.60 La. Leakage from containment isolation valves that are sealed with fluid from a seal system may be excluded when determining the combined leakage rate: *Provided*, That;

(a) Such valves have been demonstrated to have fluid leakage rates that do not exceed those specified in the technical specifications or associated bases, and

(b) The installed isolation valve seal-water system fluid inventory is sufficient to assure the sealing function for at least 30 days at a pressure of 1.10 Pa.

D. *Periodic retest schedule*—1. *Type A test*. (a) After the preoperational leakage rate tests, a set of three Type A tests shall be performed, at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year plant inservice inspections.[2]

(b) Permissible periods for testing. The performance of Type A tests shall be limited

[2] Such inservice inspections are required by §50.55a.

to periods when the plant facility is nonoperational and secured in the shutdown condition under the administrative control and in accordance with the safety procedures defined in the license.

2. *Type B tests.* (a) Type B tests, except tests for air locks, shall be performed during reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than 2 years. If opened following a Type A or B test, containment penetrations subject to Type B testing shall be Type B tested prior to returning the reactor to an operating mode requiring containment integrity. For primary reactor containment penetrations employing a continuous leakage monitoring system, Type B tests, except for tests of air locks, may, notwithstanding the test schedule specified under III.D.1., be performed every other reactor shutdown for refueling but in no case at intervals greater than 3 years.

(b)(i) Air locks shall be tested prior to initial fuel loading and at 6-month intervals thereafter at an internal pressure not less than P_a.

(ii) Air locks opened during periods when containment integrity is not required by the plant's Technical Specifications shall be tested at the end of such periods at not less than P_a.

(iii) Air locks opened during periods when containment integrity is required by the plant's Technical Specifications shall be tested within 3 days after being opened. For air lock doors opened more frequently than once every 3 days, the air lock shall be tested at least once every 3 days during the period of frequent openings. For air lock doors having testable seals, testing the seals fulfills the 3-day test requirements. In the event that the testing for this 3-day interval cannot be at P_a, the test pressure shall be as stated in the Technical Specifications. Air lock door seal testing shall not be substituted for the 6-month test of the entire air lock at not less than P_a.

(iv) The acceptance criteria for air lock testing shall be stated in the Technical Specifications.

3. *Type C tests.* Type C tests shall be performed during each reactor shutdown for refueling but in no case at intervals greater than 2 years.

IV. SPECIAL TESTING REQUIREMENTS

A. *Containment modification.* Any major modification, replacement of a component which is part of the primary reactor containment boundary, or resealing a seal-welded door, performed after the preoperational leakage rate test shall be followed by either a Type A, Type B, or Type C test, as applicable for the area affected by the modification. The measured leakage from this test shall be included in the summary report required by V.B. The acceptance criteria of III.A.5.(b), III.B.3., or III.C.3., as appropriate, shall be met. Minor modifications, replacements, or resealing of seal-welded doors, performed directly prior to the conduct of a scheduled Type A test do not require a separate test.

B. *Multiple leakage barrier or subatmospheric containments.* The primary reactor containment barrier of a multiple barrier or subatmospheric containment shall be subjected to Type A tests to verify that its leakage rate meets the requirements of this appendix. Other structures of multiple barrier or subatmospheric containments (e.g., secondary containments for boiling water reactors and shield buildings for pressurized water reactors that enclose the entire primary reactor containment or portions thereof) shall be subject to individual tests in accordance with the procedures specified in the technical specifications, or associated bases.

V. INSPECTION AND REPORTING OF TESTS

A. *Containment inspection.* A general inspection of the accessible interior and exterior surfaces of the containment structures and components shall be performed prior to any Type A test to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leak-tightness. If there is evidence of structural deterioration, Type A tests shall not be performed until corrective action is taken in accordance with repair procedures, non destructive examinations, and tests as specified in the applicable code specified in §50.55a at the commencement of repair work. Such structural deterioration and corrective actions taken shall be included in the summary report required by V.B.

B. *Recordkeeping of test results.* 1. The preoperational and periodic tests must be documented in a readily available summary report that will be made available for inspection, upon request, at the nuclear power plant. The summary report shall include a schematic arrangement of the leakage rate measurement system, the instrumentation used, the supplemental test method, and the test program selected as applicable to the preoperational test, and all the subsequent periodic tests. The report shall contain an analysis and interpretation of the leakage rate test data for the Type A test results to the extent necessary to demonstrate the acceptability of the containment's leakage rate in meeting acceptance criteria.

2. For each periodic test, leakage test results from Type A, B, and C tests shall be included in the summary report. The summary report shall contain an analysis and interpretation of the Type A test results and a summary analysis of periodic Type B and Type C tests that were performed since the last type A test. Leakage test results from type A, B, and C tests that failed to meet the acceptance criteria of III.A.5(b), III.B.3, and III.C.3, respectively, shall be included in a

separate accompanying summary report that includes an analysis and interpretation of the test data, the least squares fit analysis of the test data, the instrumentation error analysis, and the structural conditions of the containment or components, if any, which contributed to the failure in meeting the acceptance criteria. Results and analyses of the supplemental verification test employed to demonstrate the validity of the leakage rate test measurements shall also be included.

OPTION B—PERFORMANCE-BASED REQUIREMENTS

Table of Contents

I. INTRODUCTION

One of the conditions required of all operating licenses for light-water-cooled power reactors as specified in § 50.54(o) is that primary reactor containments meet the leakage-rate test requirements in either Option A or B of this appendix. These test requirements ensure that (a) leakage through these containments or systems and components penetrating these containments does not exceed allowable leakage rates specified in the Technical Specifications and (b) integrity of the containment structure is maintained during its service life. Option B of this appendix identifies the performance-based requirements and criteria for preoperational and subsequent periodic leakage-rate testing.[3]

II. DEFINITIONS

Performance criteria means the performance standards against which test results are to be compared for establishing the acceptability of the containment system as a leakage-limiting boundary.

Containment system means the principal barrier, after the reactor coolant pressure boundary, to prevent the release of quantities of radioactive material that would have a significant radiological effect on the health of the public.

Overall integrated leakage rate means the total leakage rate through all tested leakage paths, including containment welds, valves, fittings, and components that penetrate the containment system.

La (percent/24 hours) means the maximum allowable leakage rate at pressure Pa as specified in the Technical Specifications.

Pa (p.s.i.g) means the calculated peak containment internal pressure related to the design basis loss-of-coolant accident as specified in the Technical Specifications.

III. PERFORMANCE-BASED LEAKAGE-TEST REQUIREMENTS

A. Type A Test

Type A tests to measure the containment system overall integrated leakage rate must be conducted under conditions representing design basis loss-of-coolant accident containment peak pressure. A Type A test must be conducted (1) after the containment system has been completed and is ready for operation and (2) at a periodic interval based on the historical performance of the overall containment system as a barrier to fission product releases to reduce the risk from reactor accidents. A general visual inspection of the accessible interior and exterior surfaces of the containment system for structural deterioration which may affect the containment leak-tight integrity must be conducted prior to each test, and at a periodic interval between tests based on the performance of the containment system. The leakage rate must not exceed the allowable leakage rate (La) with margin, as specified in the Technical Specifications. The test results must be compared with previous results to examine the performance history of the overall containment system to limit leakage.

B. Type B and C Tests

Type B pneumatic tests to detect and measure local leakage rates across pressure retaining, leakage-limiting boundaries, and Type C pneumatic tests to measure containment isolation valve leakage rates, must be conducted (1) prior to initial criticality, and (2) periodically thereafter at intervals based on the safety significance and historical performance of each boundary and isolation valve to ensure the integrity of the overall containment system as a barrier to fission product release to reduce the risk from reactor accidents. The performance-based testing program must contain a performance criterion for Type B and C tests, consideration of leakage-rate limits and factors that are indicative of or affect performance, when establishing test intervals, evaluations of performance of containment system components, and comparison to previous test results to examine the performance history of the overall containment system to limit

[3] Specific guidance concerning a performance-based leakage-test program, acceptable leakage-rate test methods, procedures, and analyses that may be used to implement these requirements and criteria are provided in Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program.''

leakage. The tests must demonstrate that the sum of the leakage rates at accident pressure of Type B tests, and pathway leakage rates from Type C tests, is less than the performance criterion (La) with margin, as specified in the Technical Specification.

IV. RECORDKEEPING

The results of the preoperational and periodic Type A, B, and C tests must be documented to show that performance criteria for leakage have been met. The comparison to previous results of the performance of the overall containment system and of individual components within it must be documented to show that the test intervals established for the containment system and components within it are adequate. These records must be available for inspection at plant sites.

If the test results exceed the performance criteria (La) as defined in the plant Technical Specifications, those exceedances must be assessed for Emergency Notification System reporting under §§50.72 (b)(1)(ii) and §50.72 (b)(2)(i), and for a Licensee Event Report under §50.73 (a)(2)(ii).

V. APPLICATION

A. Applicability

The requirements in either or both Option B, III.A for Type A tests, and Option B, III.B for Type B and C tests, may be adopted on a voluntary basis by an operating nuclear power reactor licensee as specified in §50.54 in substitution of the requirements for those tests contained in Option A of this appendix. If the requirements for tests in Option B, III.A or Option B, III.B are implemented, the recordkeeping requirements in Option B, IV for these tests must be substituted for the reporting requirements of these tests contained in Option A of this appendix.

B. Implementation

1. Specific exemptions to Option A of this appendix that have been formally approved by the AEC or NRC, according to 10 CFR 50.12, are still applicable to Option B of this appendix if necessary, unless specifically revoked by the NRC.

2. A licensee or applicant for an operating license may adopt Option B, or parts thereof, as specified in Section V.A of this appendix, by submitting its implementation plan and request for revision to technical specifications (see paragraph B.3 below) to the Director of the Office of Nuclear Reactor Regulation.

3. The regulatory guide or other implementation document used by a licensee, or applicant for an operating license, to develop a performance-based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide.

4. The detailed licensee programs for conducting testing under Option B must be available at the plant site for NRC inspection.

[38 FR 4386, Feb. 14, 1973; 38 FR 5997, Mar. 6, 1973, as amended at 41 FR 16447, Apr. 19, 1976; 45 FR 62789, Sept. 22, 1980; 51 FR 40311, Nov. 6, 1986; 53 FR 45891, Nov. 15, 1988; 57 FR 61786, Dec. 29, 1992; 59 FR 50689, Oct. 5, 1994; 60 FR 13616, Mar. 14, 1995; 60 FR 49504, Sept. 26, 1995]

APPENDIX K TO PART 50—ECCS EVALUATION MODELS

I. Required and Acceptable Features of Evaluation Models.

II. Required Documentation.

I. REQUIRED AND ACCEPTABLE FEATURES OF THE EVALUATION MODELS

A. *Sources of heat during the LOCA*. For the heat sources listed in paragraphs I.A.1 to 4 of this appendix it must be assumed that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level (to allow for instrumentation error), with the maximum peaking factor allowed by the technical specifications. An assumed power level lower than the level specified in this paragraph (but not less than the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error. A range of power distribution shapes and peaking factors representing power distributions that may occur over the core lifetime must be studied. The selected combination of power distribution shape and peaking factor should be the one that results in the most severe calculated consequences for the spectrum of postulated breaks and single failures that are analyzed.

1. *The Initial Stored Energy in the Fuel*. The steady-state temperature distribution and stored energy in the fuel before the hypothetical accident shall be calculated for the burn-up that yields the highest calculated cladding temperature (or, optionally, the highest calculated stored energy.) To accomplish this, the thermal conductivity of the UO_2 shall be evaluated as a function of burn-up and temperature, taking into consideration differences in initial density, and the thermal conductance of the gap between the UO_2 and the cladding shall be evaluated as a function of the burn-up, taking into consideration fuel densification and expansion, the composition and pressure of the gases within the fuel rod, the initial cold gap dimension with its tolerances, and cladding creep.

2. *Fission Heat*. Fission heat shall be calculated using reactivity and reactor kinetics. Shutdown reactivities resulting from temperatures and voids shall be given their minimum plausible values, including allowance for uncertainties, for the range of power distribution shapes and peaking factors indicated to be studied above. Rod trip and insertion may be assumed if they are calculated to occur.

3. *Decay of Actinides*. The heat from the radioactive decay of actinides, including neptunium and plutonium generated during operation, as well as isotopes of uranium, shall be calculated in accordance with fuel cycle calculations and known radioactive properties. The actinide decay heat chosen shall be that appropriate for the time in the fuel cycle that yields the highest calculated fuel temperature during the LOCA.

4. *Fission Product Decay*. The heat generation rates from radioactive decay of fission products shall be assumed to be equal to 1.2 times the values for infinite operating time in the ANS Standard (Proposed American Nuclear Society Standards—''Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors.'' Approved by Subcommittee ANS–5, ANS Standards Committee, October 1971). This standard has been approved for incorporation by reference by the Director of the Federal Register. A copy of the standard is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852–2738. The fraction of the locally generated gamma energy that is deposited in the fuel (including the cladding) may be different from 1.0; the value used shall be justified by a suitable calculation.

5. *Metal—Water Reaction Rate*. The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation (Baker, L., Just, L.C., ''Studies of Metal Water Reactions at High Temperatures, III. Experimental and Theoretical Studies of the Zirconium-Water Reaction,'' ANL–6548, page 7, May 1962). This publication has been approved for incorporation by reference by the Director of the Federal Register. A copy of the publication is available for inspection at the NRC Library, 11545 Rockville Pike, Two White Flint North, Rockville, Maryland 20852–2738. The reaction shall be assumed not to be steam limited. For rods whose cladding is calculated to rupture during the LOCA, the inside of the cladding shall be assumed to react after the rupture. The calculation of the reaction rate on the inside of the cladding shall also follow the Baker-Just equation, starting at the time when the cladding is calculated to rupture, and extending around the cladding inner circumference and axially no less that 1.5 inches each way from the location of the rupture, with the reaction assumed not to be steam limited.

6. *Reactor Internals Heat Transfer*. Heat transfer from piping, vessel walls, and non-fuel internal hardware shall be taken into account.

7. *Pressurized Water Reactor Primary-to-Secondary Heat Transfer*. Heat transferred between primary and secondary systems through heat exchangers (steam generators) shall be taken into account. (Not applicable to Boiling Water Reactors.)

B. Swelling and Rupture of the Cladding and Fuel Rod Thermal Parameters

Each evaluation model shall include a provision for predicting cladding swelling and rupture from consideration of the axial temperature distribution of the cladding and from the difference in pressure between the inside and outside of the cladding, both as functions of time. To be acceptable the swelling and rupture calculations shall be based on applicable data in such a way that the degree of swelling and incidence of rupture are not underestimated. The degree of swelling and rupture shall be taken into account in calculations of gap conductance, cladding oxidation and embrittlement, and hydrogen generation.

The calculations of fuel and cladding temperatures as a function of time shall use values for gap conductance and other thermal parameters as functions of temperature and other applicable time-dependent variables. The gap conductance shall be varied in accordance with changes in gap dimensions and any other applicable variables.

C. Blowdown Phenomena

1. *Break Characteristics and Flow*. a. In analyses of hypothetical loss-of-coolant accidents, a spectrum of possible pipe breaks shall be considered. This spectrum shall include instantaneous double-ended breaks ranging in cross-sectional area up to and including that of the largest pipe in the primary coolant system. The analysis shall also include the effects of longitudinal splits in the largest pipes, with the split area equal to the cross-sectional area of the pipe.

b. *Discharge Model*. For all times after the discharging fluid has been calculated to be two-phase in composition, the discharge rate shall be calculated by use of the Moody model (F.J. Moody, ''Maximum Flow Rate of a Single Component, Two-Phase Mixture,'' Journal of Heat Transfer, Trans American Society of Mechanical Engineers, 87, No. 1, February, 1965). This publication has been approved for incorporation by reference by the Director of the Federal Register. A copy of this publication is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852–2738. The calculation shall be conducted with at least three values of a discharge coefficient applied to the postulated break area, these values spanning the range from 0.6 to 1.0. If the results indicate that the maximum clad temperature for the hypothetical accident is to be found at an even lower value of the discharge coefficient, the range of discharge coefficients shall be extended until the maximum clad temperatures calculated by this variation has been achieved.

c. *End of Blowdown.* (Applies Only to Pressurized Water Reactors.) For postulated cold leg breaks, all emergency cooling water injected into the inlet lines or the reactor vessel during the bypass period shall in the calculations be subtracted from the reactor vessel calculated inventory. This may be executed in the calculation during the bypass period, or as an alternative the amount of emergency core cooling water calculated to be injected during the bypass period may be subtracted later in the calculation from the water remaining in the inlet lines, downcomer, and reactor vessel lower plenum after the bypass period. This bypassing shall end in the calculation at a time designated as the ''end of bypass,'' after which the expulsion or entrainment mechanisms responsible for the bypassing are calculated not to be effective. The end-of-bypass definition used in the calculation shall be justified by a suitable combination of analysis and experimental data. Acceptable methods for defining ''end of bypass'' include, but are not limited to, the following: (1) Prediction of the blowdown calculation of downward flow in the downcomer for the remainder of the blowdown period; (2) Prediction of a threshold for droplet entrainment in the upward velocity, using local fluid conditions and a conservative critical Weber number.

d. *Noding Near the Break and the ECCS Injection Points.* The noding in the vicinity of and including the broken or split sections of pipe and the points of ECCS injection shall be chosen to permit a reliable analysis of the thermodynamic history in these regions during blowdown.

2. *Frictional Pressure Drops.* The frictional losses in pipes and other components including the reactor core shall be calculated using models that include realistic variation of friction factor with Reynolds number, and realistic two-phase friction multipliers that have been adequately verified by comparison with experimental data, or models that prove at least equally conservative with respect to maximum clad temperature calculated during the hypothetical accident. The modified Baroczy correlation (Baroczy, C. J., ''A Systematic Correlation for Two-Phase Pressure Drop,'' *Chem. Enging. Prog. Symp. Series,* No. 64, Vol. 62, 1965) or a combination of the Thom correlation (Thom, J.R.S., ''Prediction of Pressure Drop During Forced Circulation Boiling of Water,'' *Int. J. of Heat & Mass Transfer,* 7, 709–724, 1964) for pressures equal to or greater than 250 psia and the Martinelli-Nelson correlation (Martinelli, R. C. Nelson, D.B., ''Prediction of Pressure Drop During Forced Circulation Boiling of Water,'' *Transactions of ASME,* 695–702, 1948) for pressures lower than 250 psia is acceptable as a basis for calculating realistic two-phase friction multipliers.

3. *Momentum Equation.* The following effects shall be taken into account in the conservation of momentum equation: (1) temporal change of momentum, (2) momentum convection, (3) area change momentum flux, (4) momentum change due to compressibility, (5) pressure loss resulting from wall friction, (6) pressure loss resulting from area change, and (7) gravitational acceleration. Any omission of one or more of these terms under stated circumstances shall be justified by comparative analyses or by experimental data.

4. *Critical Heat Flux.* a. Correlations developed from appropriate steady-state and transient-state experimental data are acceptable for use in predicting the critical heat flux (CHF) during LOCA transients. The computer programs in which these correlations are used shall contain suitable checks to assure that the physical parameters are within the range of parameters specified for use of the correlations by their respective authors.

b. Steady-state CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following:

(1) *W 3.* L. S. Tong, ''Prediction of Departure from Nucleate Boiling for an Axially Non-uniform Heat Flux Distribution,'' *Journal of Nuclear Energy,* Vol. 21, 241–248, 1967.

(2) *B&W–2.* J. S. Gellerstedt, R. A. Lee, W. J. Oberjohn, R. H. Wilson, L. J. Stanek, ''Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water,'' *Two-Phase Flow and Heat Transfer in Rod Bundles,* ASME, New York, 1969.

(3) *Hench-Levy.* J. M. Healzer, J. E. Hench, E. Janssen, S. Levy, ''Design Basis for Critical Heat Flux Condition in Boiling Water Reactors,'' APED–5186, GE Company Private report, July 1966.

(4) *Macbeth.* R. V. Macbeth, ''An Appraisal of Forced Convection Burnout Data,'' *Proceedings of the Institute of Mechanical Engineers,* 1965–1966.

(5) *Barnett.* P. G. Barnett, ''A Correlation of Burnout Data for Uniformly Heated Annuli and Its Uses for Predicting Burnout in Uniformly Heated Rod Bundles,'' AEEW-R 463, 1966.

(6) *Hughes.* E. D. Hughes, ''A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia,'' IN–1412, Idaho Nuclear Corporation, July 1970.

c. Correlations of appropriate transient CHF data may be accepted for use in LOCA transient analyses if comparisons between the data and the correlations are provided to demonstrate that the correlations predict values of CHF which allow for uncertainty in the experimental data throughout the range of parameters for which the correlations are to be used. Where appropriate, the comparisons shall use statistical uncertainty analysis of the data to demonstrate the conservatism of the transient correlation.

d. Transient CHF correlations acceptable for use in LOCA transients include, but are not limited to, the following:

(1) *GE transient CHF*. B. C. Slifer, J. E. Hench, ''Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors,'' NEDO–10329, General Electric Company, Equation C–32, April 1971.

e. After CHF is first predicted at an axial fuel rod location during blowdown, the calculation shall not use nucleate boiling heat transfer correlations at that location subsequently during the blowdown even if the calculated local fluid and surface conditions would apparently justify the reestablishment of nucleate boiling. Heat transfer assumptions characteristic of return to nucleate boiling (rewetting) shall be permitted when justified by the calculated local fluid and surface conditions during the reflood portion of a LOCA.

5. *Post-CHF Heat Transfer Correlations*. a. Correlations of heat transfer from the fuel cladding to the surrounding fluid in the post-CHF regimes of transition and film boiling shall be compared to applicable steady-state and transient-state data using statistical correlation and uncertainty analyses. Such comparison shall demonstrate that the correlations predict values of heat transfer coefficient equal to or less than the mean value of the applicable experimental heat transfer data throughout the range of parameters for which the correlations are to be used. The comparisons shall quantify the relation of the correlations to the statistical uncertainty of the applicable data.

b. The Groeneveld flow film boiling correlation (equation 5.7 of D.C. Groeneveld, ''An Investigation of Heat Transfer in the Liquid Deficient Regime,'' AECL–3281, revised December 1969) and the Westinghouse correlation of steady-state transition boiling (''Proprietary Redirect/Rebuttal Testimony of Westinghouse Electric Corporation,'' USNRC Docket RM–50–1, page 25–1, October 26, 1972) are acceptable for use in the post-CHF boiling regimes. In addition, the transition boiling correlation of McDonough, Milich, and King (J.B. McDonough, W. Milich, E.C. King, ''An Experimental Study of Partial Film Boiling Region with Water at Elevated Pressures in a Round Vertical Tube,'' Chemical Engineering Progress Symposium Series, Vol. 57, No. 32, pages 197–208, (1961) is suitable for use between nucleate and film boiling. Use of all these correlations is restricted as follows:

(1) The Groeneveld correlation shall not be used in the region near its low-pressure singularity,

(2) The first term (nucleate) of the Westinghouse correlation and the entire McDonough, Milich, and King correlation shall not be used during the blowdown after the temperature difference between the clad and the saturated fluid first exceeds 300 °F,

(3) Transition boiling heat transfer shall not be reapplied for the remainder of the LOCA blowdown, even if the clad superheat returns below 300 °F, except for the reflood portion of the LOCA when justified by the calculated local fluid and surface conditions.

c. Evaluation models approved after October 17, 1988, which make use of the Dougall-Rohsenow flow film boiling correlation (R.S. Dougall and W.M. Rohsenow, ''Film Boiling on the Inside of Vertical Tubes with Upward Flow of Fluid at Low Qualities,'' MIT Report Number 9079 26, Cambridge, Massachusetts, September 1963) may not use this correlation under conditions where nonconservative predictions of heat transfer result. Evaluation models that make use of the Dougall-Rohsenow correlation and were approved prior to October 17, 1988, continue to be acceptable until a change is made to, or an error is corrected in, the evaluation model that results in a significant reduction in the overall conservatism in the evaluation model. At that time continued use of the Dougall-Rohsenow correlation under conditions where nonconservative predictions of heat transfer result will no longer be acceptable. For this purpose, a significant reduction in the overall conservatism in the evaluation model would be a reduction in the calculated peak fuel cladding temperature of at least 50 °F from that which would have been calculated on October 17, 1988, due either to individual changes or error corrections or the net effect of an accumulation of changes or error corrections.

6. *Pump Modeling*. The characteristics of rotating primary system pumps (axial flow, turbine, or centrifugal) shall be derived from a dynamic model that includes momentum transfer between the fluid and the rotating member, with variable pump speed as a function of time. The pump model resistance used for analysis should be justified. The pump model for the two-phase region shall be verified by applicable two-phase pump performance data. For BWR's after saturation is calculated at the pump suction, the pump head may be assumed to vary linearly with quality, going to zero for one percent quality at the pump suction, so long as the analysis shows that core flow stops before the quality at pump suction reaches one percent.

7. *Core Flow Distribution During Blowdown*. (Applies only to pressurized water reactors.)

a. The flow rate through the hot region of the core during blowdown shall be calculated as a function of time. For the purpose of these calculations the hot region chosen shall not be greater than the size of one fuel assembly. Calculations of average flow and flow in the hot region shall take into account cross flow between regions and any flow blockage calculated to occur during blowdown as a result of cladding swelling or rupture. The calculated flow shall be smoothed to eliminate any calculated rapid oscillations (period less than 0.1 seconds).

b. A method shall be specified for determining the enthalpy to be used as input data to the hot channel heatup analysis from quantities calculated in the blowdown analysis, consistent with the flow distribution calculations.

D. Post-Blowdown Phenomena; Heat Removal by the ECCS

1. *Single Failure Criterion.* An analysis of possible failure modes of ECCS equipment and of their effects on ECCS performance must be made. In carrying out the accident evaluation the combination of ECCS subsystems assumed to be operative shall be those available after the most damaging single failure of ECCS equipment has taken place.

2. *Containment Pressure.* The containment pressure used for evaluating cooling effectiveness during reflood and spray cooling shall not exceed a pressure calculated conservatively for this purpose. The calculation shall include the effects of operation of all installed pressure-reducing systems and processes.

3. *Calculation of Reflood Rate for Pressurized Water Reactors.* The refilling of the reactor vessel and the time and rate of reflooding of the core shall be calculated by an acceptable model that takes into consideration the thermal and hydraulic characteristics of the core and of the reactor system. The primary system coolant pumps shall be assumed to have locked impellers if this assumption leads to the maximum calculated cladding temperature; otherwise the pump rotor shall be assumed to be running free. The ratio of the total fluid flow at the core exit plane to the total liquid flow at the core inlet plane (carryover fraction) shall be used to determine the core exit flow and shall be determined in accordance with applicable experimental data (for example, ''PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report,'' Westinghouse Report WCAP–7665, April 1971; ''PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report,'' Westinghouse Report WCAP–7435, January 1970; ''PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Group II Test Report,'' Westinghouse Report WCAP–7544, September 1970; ''PWR FLECHT Final Report Supplement,'' Westinghouse Report WCAP–7931, October 1972).

The effects on reflooding rate of the compressed gas in the accumulator which is discharged following accumulator water discharge shall also be taken into account.

4. *Steam Interaction with Emergency Core Cooling Water in Pressurized Water Reactors.* The thermal-hydraulic interaction between steam and all emergency core cooling water shall be taken into account in calculating the core reflooding rate. During refill and reflood, the calculated steam flow in unbroken reactor coolant pipes shall be taken to be zero during the time that accumulators are discharging water into those pipes unless experimental evidence is available regarding the realistic thermal-hydraulic interaction between the steam and the liquid. In this case, the experimental data may be used to support an alternate assumption.

5. *Refill and Reflood Heat Transfer for Pressurized Water Reactors.* a. For reflood rates of one inch per second or higher, reflood heat transfer coefficients shall be based on applicable experimental data for unblocked cores including FLECHT results (''PWR FLECHT (Full Length Emergency Cooling Heat Transfer) Final Report,'' Westinghouse Report WCAP–7665, April 1971). The use of a correlation derived from FLECHT data shall be demonstrated to be conservative for the transient to which it is applied; presently available FLECHT heat transfer correlations (''PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report,'' Westinghouse Report WCAP–7544, September 1970; ''PWR FLECHT Final Report Supplement,'' Westinghouse Report WCAP–7931, October 1972) are not acceptable. Westinghouse Report WCAP–7665 has been approved for incorporation by reference by the Director of the Federal Register. A copy of this report is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852–2738. New correlations or modifications to the FLECHT heat transfer correlations are acceptable only after they are demonstrated to be conservative, by comparison with FLECHT data, for a range of parameters consistent with the transient to which they are applied.

b. During refill and during reflood when reflood rates are less than one inch per second, heat transfer calculations shall be based on the assumption that cooling is only by steam, and shall take into account any flow blockage calculated to occur as a result of cladding swelling or rupture as such blockage might affect both local steam flow and heat transfer.

6. *Convective Heat Transfer Coefficients for Boiling Water Reactor Fuel Rods Under Spray Cooling.* Following the blowdown period, convective heat transfer shall be calculated using coefficients based on appropriate experimental data. For reactors with jet pumps and having fuel rods in a 7×7 fuel assembly

array, the following convective coefficients are acceptable:

a. During the period following lower plenum flashing but prior to the core spray reaching rated flow, a convective heat transfer coefficient of zero shall be applied to all fuel rods.

b. During the period after core spray reaches rated flow but prior to reflooding, convective heat transfer coefficients of 3.0, 3.5, 1.5, and 1.5 Btu-hr^{-1}-ft^{-2} °F^{-1} shall be applied to the fuel rods in the outer corners, outer row, next to outer row, and to those remaining in the interior, respectively, of the assembly.

c. After the two-phase reflooding fluid reaches the level under consideration, a convective heat transfer coefficient of 25 Btu-hr^{-1}-ft^{-2} °F^{-1} shall be applied to all fuel rods.

7. *The Boiling Water Reactor Channel Box Under Spray Cooling*. Following the blowdown period, heat transfer from, and wetting of, the channel box shall be based on appropriate experimental data. For reactors with jet pumps and fuel rods in a 7×7 fuel assembly array, the following heat transfer coefficients and wetting time correlation are acceptable.

a. During the period after lower plenum flashing, but prior to core spray reaching rated flow, a convective coefficient of zero shall be applied to the fuel assembly channel box.

b. During the period after core spray reaches rated flow, but prior to wetting of the channel, a convective heat transfer coefficient of 5 Btu-hr^{-1}-ft^{-2}- °F^{-1} shall be applied to both sides of the channel box.

c. Wetting of the channel box shall be assumed to occur 60 seconds after the time determined using the correlation based on the Yamanouchi analysis (''Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors,'' General Electric Company Report NEDO–10329, April 1971). This report was approved for incorporation by reference by the Director of the Federal Register. A copy of the report is available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Maryland 20852–2738.

II. Required Documentation

1. a. A description of each evaluation model shall be furnished. The description shall be sufficiently complete to permit technical review of the analytical approach including the equations used, their approximations in difference form, the assumptions made, and the values of all parameters or the procedure for their selection, as for example, in accordance with a specified physical law or empirical correlation.

b. A complete listing of each computer program, in the same form as used in the evaluation model, must be furnished to the Nuclear Regulatory Commission upon request.

2. For each computer program, solution convergence shall be demonstrated by studies of system modeling or noding and calculational time steps.

3. Appropriate sensitivity studies shall be performed for each evaluation model, to evaluate the effect on the calculated results of variations in noding, phenomena assumed in the calculation to predominate, including pump operation or locking, and values of parameters over their applicable ranges. For items to which results are shown to be sensitive, the choices made shall be justified.

4. To the extent practicable, predictions of the evaluation model, or portions thereof, shall be compared with applicable experimental information.

5. General Standards for Acceptability—Elements of evaluation models reviewed will include technical adequacy of the calculational methods, including: For models covered by §50.46(a)(1)(ii), compliance with required features of section I of this appendix K; and, for models covered by §50.46(a)(1)(i), assurance of a high level of probability that the performance criteria of §50.46(b) would not be exceeded.

[39 FR 1003, Jan. 4, 1974, as amended at 51 FR 40311, Nov. 6, 1986; 53 FR 36005, Sept. 16, 1988; 57 FR 61786, Dec. 29, 1992; 59 FR 50689, Oct. 5, 1994; 60 FR 24552, May 9, 1995; 65 FR 34921, June 1, 2000]

Appendix L to Part 50 [Reserved]

Appendix M to Part 50—Standardization of Design; Manufacture of Nuclear Power Reactors; Construction and Operation of Nuclear Power Reactors Manufactured Pursuant to Commission License

Section 101 of the Atomic Energy Act of 1954, as amended, and §50.10 require a Commission license to transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, use, import, or export any production or utilization facility. The regulations in the part require the issuance of a construction permit by the Commission before commencement of construction of a production or utilization facility, and the issuance of an operating license before operation of the facility. The provisions of this part relating to the facility licensing process are, in general, predicated on the assumption that the facility will be assembled and constructed on the site at which it is to be operated. In those circumstances, both facility design and site-related issues can be considered in the initial, construction permit stage of the licensing process.

However, under the Atomic Energy Act, a license may be sought and issued authorizing the manufacture of facilities but not their construction and installation at the sites on which the facilities are to be operated. Prior to the ''commencement of construction'', as defined in §50.10(c), of a facility (manufactured pursuant to such a Commission license) on the site at which it is to operate—that is preparation of the site and installation of the facility—a construction permit that, among other things, reflects approval of the site on which the facility is to be operated, must be issued by the Commission. This appendix sets out the particular requirements and provisions applicable to such situations where nuclear power reactors to be manufactured pursuant to a Commission license and subsequently installed at the site pursuant to a Commission construction permit, are of the type described in §50.22. It thus codifies one approach to the standardization of nuclear power reactors.

1. Except as otherwise specified in this appendix or as the context otherwise indicates, the provisions in this part applicable to construction permits, including the requirement in §50.58 for review of the application by the Advisory Committee on Reactor Safeguards and the holding of a public hearing, apply in context, with respect to matters of radiological health and safety, environmental protection, and the common defense and security, to licenses pursuant to this appendix M to manufacture nuclear power reactors (manufacturing licenses) to be operated at sites not identified in the license application.

2. An application for a manufacturing license pursuant to this appendix M must be submitted, as specified in §50.4, and meet all the requirements of §§50.34(a) (1)–(9) and 50.34a (a) and (b), except that the preliminary safety analysis report shall be designated as a ''design report'' and any required information or analyses relating to site matters shall be predicated on postulated site parameters which must be specified in the application. The application must also include information pertaining to design features of the proposed reactor(s) that affect plans for coping with emergencies in the operation of the reactor(s).

3. An applicant for a manufacturing license pursuant to this appendix M shall submit with his application an environmental report as required of applicants for construction permits in accordance with subpart A of part 51 of this chapter, provided, however, that such report shall be directed at the manufacture of the reactor(s) at the manufacturing site; and, in general terms, at the construction and operation of the reactor(s) at a hypothetical site or sites having characteristics that fall within the postulated site parameters. The related draft and final environmental impact statement prepared by the Commission's regulatory staff will be similarly directed.

4. (a) Sections 50.10 (b) and (c), 50.12(b), 50.23, 50.30(d), 50.34(a)(10), 50.34a(c), 50.35 (a) and (c), 50.40(a), 50.45, 50.55(d), 50.56, and appendix J do not apply to manufacturing licenses. Appendices E and H apply to manufacturing licenses only to the extent that the requirements of these appendixes involve facility design features.

(b) The financial information submitted pursuant to §50.33(f) and appendix C shall be directed at a demonstration of the financial qualifications of the applicant for the manufacturing license to carry out the manufacturing activity for which the license is sought.

5. The Commission may issue a license to manufacture one or more nuclear power reactors to be operated at sites not identified in the license application if the Commission finds that:

(a) The applicant has described the proposed design of and the site parameters postulated for the reactor(s), including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public.

(b) Such further technical or design information as may be required to complete the design report and which can reasonably be left for later consideration, will be supplied in a supplement to the design report.

(c) Safety features or components, if any, which require research and development have been described by the applicant and the applicant has identified, and there will be conducted a research and development program reasonably designed to resolve any safety questions associated with such features or components; and

(d) On the basis of the foregoing, there is reasonable assurance that (i) such safety questions will be satisfactorily resolved before any of the proposed nuclear power reactor(s) are removed from the manufacturing site and (ii) taking into consideration the site criteria contained in part 100 of this chapter, the proposed reactor(s) can be constructed and operated at sites having characteristics that fall within the site parameters postulated for the design of the reactor(s) without undue risk to the health and safety of the public.

(e) The applicant is technically and financially qualified to design and manufacture the proposed nuclear power reactor(s).

(f) The issuance of a license to the applicant will not be inimical to the common defense and security or to the health and safety of the public.

(g) On the basis of the evaluations and analyses of the environmental effects of the proposed action required by subpart A of part 51 of this chapter and paragraph 3 of this appendix, the action called for is the issuance of the license.

NOTE: When an applicant has supplied initially all of the technical information required to complete the application, including the final design of the reactor(s), the findings required for the issuance of the license will be appropriately modified to reflect that fact.

6. Each manufacturing license issued pursuant to this appendix will specify the number of nuclear power reactors authorized to be manufactured and the latest date for the completion of the manufacture of all such reactors. Upon good cause shown, the Commission will extend such completion date for a reasonable period of time.

7. The holder of a manufacturing license issued pursuant to this appendix M shall submit to the Commission the final design of the nuclear power reactor(s) covered by the license as soon as such design has been completed. Such submittal shall be in the form of an application for amendment of the manufacturing license.

8. The prohibition in § 50.10(c) against commencement of construction of a production or utilization facility prior to issuance of a construction permit applies to the transport of a nuclear power reactor(s) manufactured pursuant to a license issued pursuant to this appendix from the manufacturing facility to the site at which the reactor(s) will be installed and operated. In addition, such nuclear power reactor(s) shall not be removed from the manufacturing site until the final design of the reactor(s) has been approved by the Commission in accordance with paragraph 7.

9. An application for a permit to construct a nuclear power reactor(s) which is the subject of an application for a manufacturing license pursuant to this appendix M need not contain such information or analyses as have previously been submitted to the Commission in connection with the application for a manufacturing license, but shall contain, in addition to the information and analyses otherwise required by §§ 50.34(a) and 50.34a, sufficient information to demonstrate that the site on which the reactor(s) is to be operated falls within the postulated site parameters specified in the relevant manufacturing license application.

10. The Commission may issue a permit to construct a nuclear power reactor(s) which is the subject of an application for a manufacturing license pursuant to this appendix M if the Commission (a) finds that the site on which the reactor is to be operated falls within the postulated site parameters specified in the relevant application for a manufacturing license and (b) makes the findings otherwise required by this part. In no event will a construction permit be issued until the relevant manufacturing license has been issued.

11. An operating license for a nuclear power reactor(s) that has been manufactured under a Commission license issued pursuant to this appendix M may be issued by the Commission pursuant to § 50.57 and subpart A of part 51 of this chapter except that the Commission shall find, pursuant to § 50.57(a)(1), that construction of the reactor(s) has been substantially completed in conformity with both the manufacturing license and the construction permit and the applications therefor, as amended, and the provisions of the Act, and the rules and regulations of the Commission. Notwithstanding the other provisions of this paragraph, no application for an operating license for a nuclear power reactor(s) that has been manufactured under a Commission license issued pursuant to this appendix M will be docketed until the application for an amendment to the relevant manufacturing license required by paragraph 7 has been docketed.

12. In making the findings required by this part for the issuance of a construction permit or an operating license for a nuclear power reactor(s) that has been manufactured under a Commission license issued pursuant to this appendix, or an amendment to such a manufacturing license, construction permit, or operating license, the Commission will treat as resolved those matters which have been resolved at an earlier stage of the licensing process, unless there exists significant new information that substantially affects the conclusion(s) reached at the earlier stage or other good cause.

[38 FR 30253, Nov. 2, 1973, as amended at 49 FR 9404, Mar. 12, 1984; 49 FR 35754, Sept. 12, 1984; 50 FR 18853, May 3, 1985; 51 FR 40311, Nov. 6, 1986]

APPENDIX N TO PART 50—STANDARDIZATION OF NUCLEAR POWER PLANT DESIGNS: LICENSES TO CONSTRUCT AND OPERATE NUCLEAR POWER REACTORS OF DUPLICATE DESIGN AT MULTIPLE SITES

Section 101 of the Atomic Energy Act of 1954, as amended, and § 50.10 of this part require a Commission license to transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, use, import or export any production or utilization facility. The regulations in this part require the issuance of a construction permit by the Commission before commencement of construction of a production or utilization facility, except as provided in § 50.10(e), and the issuance of an operating license before operation of the facility.

The Commission's regulations in part 2 of this chapter specifically provide for the holding of hearings on particular issues separately from other issues involved in hearings in licensing proceedings (§ 2.761a, appendix A, section I(c)), and for the consolidation of adjudicatory proceedings and of the presentations of parties in adjudicatory proceedings such as licensing proceedings (§§2.715a, 2.716).

This appendix sets out the particular requirements and provisions applicable to situations in which applications are filed by one or more applicants for licenses to construct and operate nuclear power reactors of essentially the same design to be located at different sites.[1]

1. Except as otherwise specified in this appendix or as the context otherwise indicates, the provisions of this part applicable to construction permits and operating licenses, including the requirement in §50.58 for review of the application by the Advisory Committee on Reactor Safeguards and the holding of public hearings, apply to construction permits and operating licenses subject to this appendix N.

2. Applications for construction permits submitted pursuant to this appendix must include the information required by §§50.33, 50.34(a) and 50.34a(a) and (b) and be submitted as specified in §50.4. The applicant shall also submit the information required by §51.50 of this chapter.

3. Applications for operating licenses submitted pursuant to this appendix N shall include the information required by §§50.33, 50.34(b) and (c), and 50.34a(c). The applicant shall also submit the information required by §51.53 of this chapter. For the technical information required by §§50.34(b)(2) through (5) and 50.34a(c), reference may be made to a single final safety analysis of the design.

[40 FR 2977, Jan. 17, 1975, as amended at 49 FR 9405, Mar. 12, 1984; 51 FR 40311, Nov. 6, 1986; 70 FR 61888, Oct. 27, 2005]

Appendix O to Part 50—Standardization of Design: Staff Review of Standard Designs

This appendix sets out procedures for the filing, staff review and referral to the Advisory Committee on Reactor Safeguards of standard designs for a nuclear power reactor of the type described in §50.22 or major portions thereof.

1. Any person may submit a proposed preliminary or final standard design for a nuclear power reactor of the type described in §50.22 to the regulatory staff for its review. Such a submittal may consist of either the preliminary or final design for the entire reactor facility or the preliminary or final design of major portions thereof.

2. The submittal for review of the standard design must be made in the same manner and in the same number of copies as provided in §§50.4 and 50.30 for license applications.

3. The submittal for review of the standard design shall include the information described in §50.33(a) through (d) and the applicable technical information required by §§50.34 (a) and (b), as appropriate, and 50.34a (other than that required by §§50.34(a) (6) and (10), 50.34(b)(1), (6)(i), (ii), (iv), and (v) and 50.34(b) (7) and (8)). The submittal shall also include a description, analysis and evaluation of the interfaces between the submitted design and the balance of the nuclear power plant. With respect to the requirements of §§50.34(a)(1), the submittal for review of a standard design shall include the site parameters postulated for the design, and an analysis and evaluation of the design in terms of such postulated site parameters. The information submitted pursuant to §50.34(a)(7) shall be limited to the quality assurance program to be applied to the design, procurement and fabrication of the structures, systems, and components for which design review has been requested and the information submitted pursuant to §50.34(a)(9) shall be limited to the qualifications of the person submitting the standard design to design the reactor or major portion thereof. The submittal shall also include information pertaining to design features that affect plans for coping with emergencies in the operation of the reactor or major portion thereof.

4. Once the regulatory staff has initiated a technical review of a submittal under this appendix, the submittal will be referred to the Advisory Committee on Reactor Safeguards (ACRS) for a review and report.

5. Upon completion of their review of a submittal under this appendix, the NRC regulatory staff shall publish in the Federal Register a determination as to whether or not the preliminary or final design is acceptable, subject to such conditions as may be appropriate, and make available at the NRC Web site, *http://www.nrc.gov*, an analysis of the design in the form of a report. An approved design shall be utilized by and relied upon by the regulatory staff and the ACRS in their review of any individual facility license application which incorporates by reference a design approved in accordance with this paragraph unless there exists significant new information which substantially affects the earlier determination or other good cause.

6. The determination and report by the regulatory staff shall not constitute a commitment to issue a permit or license, or in any way affect the authority of the Commission, Atomic Safety and Licensing Appeal Panel, Atomic Safety and Licensing Panel, and other presiding officers in any proceeding under subpart G of part 2 of this chapter.

[1] If the design for the power reactor(s) proposed in a particular application is not identical to the others, that application may not be processed under this appendix and subpart D of part 2 of this chapter.

7. The Commission may, on its own initiative or in response to a petition for rule making, approve the design in a rulemaking proceeding and in that event, the approved design will be subject to challenge only as provided in §2.758 of this chapter. An environmental impact statement may be prepared for such a rule making action in accordance with §§51.20(b)(13) and 51.85 of this chapter. If an environmental impact statement is prepared, the Commission may require the petitioner for rulemaking to submit information to the Commission to aid the Commission in the preparation of the environmental impact statement.

8. Information requests to the approval holder regarding an approved design shall be evaluated prior to issuance to ensure that the burden to be imposed on respondents is justified in view of the potential safety significance of the issue to be addressed in the requested information. Each such evaluation performed by the NRC staff shall be in accordance with 10 CFR 50.54(f) and shall be approved by the Executive Director for Operations or his or her designee prior to issuance of the request.

[40 FR 2977, Jan. 17, 1975, as amended at 49 FR 9405, Mar. 12, 1984; 50 FR 38112, Sept. 20, 1985; 51 FR 40311, Nov. 6, 1986; 64 FR 48952, Sept. 9, 1999]

APPENDIX P TO PART 50 [RESERVED]

APPENDIX Q TO PART 50—PRE-APPLICATION EARLY REVIEW OF SITE SUITABILITY ISSUES

This appendix sets out procedures for the filing, Staff review, and referral to the Advisory Committee on Reactor Safeguards of requests for early review of one or more site suitability issues relating to the construction and operation of certain utilization facilities separately from and prior to the submittal of applications for construction permits for the facilities. The appendix also sets out procedures for the preparation and issuance of Staff Site Reports and for their incorporation by reference in applications for the construction and operation of certain utilization facilities. The utilization facilities are those which are subject to §51.20(b) of this chapter and are of the type specified in §50.21(b) (2) or (3) or §50.22 or are testing facilities. This appendix does not apply to proceedings conducted pursuant to subpart F of part 2 of this chapter.

1. Any person may submit information regarding one or more site suitability issues to the Commission's Staff for its review separately from and prior to an application for a construction permit for a facility. Such a submittal shall be accompanied by any fee required by part 170 of this chapter and shall consist of the portion of the information required of applicants for construction permits by §§50.33(a)–(c) and (e), and, insofar as it relates to the issue(s) of site suitability for which early review is sought, by §§50.34(a)(1) and 50.30(f), except that information with respect to operation of the facility at the projected initial power level need not be supplied.

2. The submittal for early review of site suitability issue(s) must be made in the same manner and in the same number of copies as provided in §§50.4 and 50.30 for license applications. The submittal must include sufficient information concerning a range of postulated facility design and operation parameters to enable the Staff to perform the requested review of site suitability issues. The submittal must contain suggested conclusions on the issues of site suitability submitted for review and must be accompanied by a statement of the bases or the reasons for those conclusions. The submittal must also list, to the extent possible, any long-range objectives for ultimate development of the site, state whether any site selection process was used in preparing the submittal, describe any site selection process used, and explain what consideration, if any, was given to alternative sites.

3. The Staff shall publish a notice of docketing of the submittal in the FEDERAL REGISTER, and shall send a copy of the notice of docketing to the Governor or other appropriate official of the State in which the site is located. This notice shall identify the location of the site, briefly describe the site suitability issue(s) under review, and invite comments from Federal, State, and local agencies and interested persons within 120 days of publication or such other time as may be specified, for consideration by the staff in connection with the initiation or outcome of the review and, if appropriate by the ACRS, in connection with the outcome of their review. The person requesting review shall serve a copy of the submittal on the Governor or other appropriate official of the State in which the site is located, and on the chief executive of the municipality in which the site is located or, if the site is not located in a municipality, on the chief executive of the county. The portion of the submittal containing information required of applicants for construction permits by §§50.33(a)–(c) and (e) and 50.34(a)(1) will be referred to the Advisory Committee on Reactor Safeguards (ACRS) for a review and report. There will be no referral to the ACRS unless early review of the site safety issues under §50.34(a)(1) is requested.

4. Upon completion of review by the NRC staff and, if appropriate by the ACRS, of a submittal under this appendix, the NRC staff shall prepare a Staff Site Report which shall identify the location of the site, state the site suitability issues reviewed, explain the nature and scope of the review, state the

conclusions of the staff regarding the issues reviewed and state the reasons for those conclusions. Upon issuance of an NRC Staff Site Report, the NRC staff shall publish a notice of the availability of the report in the FEDERAL REGISTER and shall make the report available at the NRC Web site, *http://www.nrc.gov*. The NRC staff shall also send a copy of the report to the Governor or other appropriate official of the State in which the site is located, and to the chief executive of the municipality in which the site is located or, if the site is not located in a municipality, to the chief executive of the county.

5. Any Staff Site Report prepared and issued in accordance with this appendix may be incorporated by reference, as appropriate, in an application for a construction permit for a utilization facility which is subject to §51.20(b) of this chapter and is of the type specified in §50.21(b) (2) or (3) or §50.22 of this chapter or is a testing facility. The conclusions of the Staff Site Report will be reexamined by the staff where five years or more have elapsed between the issuance of the Staff Site Report and its incorporation by reference in a construction permit application.

6. Issuance of a Staff Site Report shall not constitute a commitment to issue a permit or license, to permit on-site work under §50.10(e), or in any way affect the authority of the Commission, Atomic Safety and Licensing Appeal Panel, Atomic Safety and Licensing Board, and other presiding officers in any proceeding under subpart F and/or G of part 2 of this chapter.

7. The staff will not conduct more than one review of site suitability issues with regard to a particular site prior to the full construction permit review required by subpart A of part 51 of this chapter. The staff may decline to prepare and issue a Staff Site Report in response to a submittal under this appendix where it appears that, (a) in cases where no review of the relative merits of the submitted site and alternative sites under subpart A of part 51 of this chapter is requested, there is a reasonable likelihood that further Staff review would identify one or more preferable alternative sites and the Staff review of one or more site suitability issues would lead to an irreversible and irretrievable commitment of resources prior to the submittal of the analysis of alternative sites in the Environmental Report that would prejudice the later review and decision on alternative sites under subpart F and/or G of part 2 and subpart A of part 51 of this chapter; or (b) in cases where, in the judgment of the Staff, early review of any site suitability issue or issues would not be in the public interest, considering (1) the degree of likelihood that any early findings on those issues would retain their validity in later reviews, (2) the objections, if any, of cognizant state or local government agencies to the conduct of an early review on those issues, and (3) the possible effect on the public interest of having an early, if not necessarily conclusive, resolution of those issues.

[42 FR 22887, May 5, 1977, as amended at 49 FR 9405, Mar. 12, 1984; 51 FR 40311, Nov. 6, 1986; 53 FR 43420, Oct. 27, 1988; 64 FR 48952, Sept. 9, 1999]

APPENDIX R TO PART 50—FIRE PROTECTION PROGRAM FOR NUCLEAR POWER FACILITIES OPERATING PRIOR TO JANUARY 1, 1979

I. INTRODUCTION AND SCOPE

This appendix applies to licensed nuclear power electric generating stations that were operating prior to January 1, 1979, except to the extent set forth in §50.48(b) of this part. With respect to certain generic issues for such facilities it sets forth fire protection features required to satisfy Criterion 3 of appendix A to this part.

Criterion 3 of appendix A to this part specifies that ''Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.''

When considering the effects of fire, those systems associated with achieving and maintaining safe shutdown conditions assume major importance to safety because damage to them can lead to core damage resulting from loss of coolant through boiloff.

The phrases ''important to safety,'' or ''safety-related,'' will be used throughout this appendix R as applying to all safety functions. The phrase ''safe shutdown'' will be used throughout this appendix as applying to both hot and cold shutdown functions.

Because fire may affect safe shutdown systems and because the loss of function of systems used to mitigate the consequences of design basis accidents under postfire conditions does not per se impact public safety, the need to limit fire damage to systems required to achieve and maintain safe shutdown conditions is greater than the need to limit fire damage to those systems required to mitigate the consequences of design basis accidents. Three levels of fire damage limits are established according to the safety functions of the structure, system, or component:

Safety function	Fire damage limits
Hot Shutdown	One train of equipment necessary to achieve hot shutdown from either the control room or emergency control station(s) must be maintained free of fire damage by a single fire, including an exposure fire.[1]

Safety function	Fire damage limits
Cold Shutdown	Both trains of equipment necessary to achieve cold shutdown may be damaged by a single fire, including an exposure fire, but damage must be limited so that at least one train can be repaired or made operable within 72 hours using onsite capability.
Design Basis Accidents.	Both trains of equipment necessary for mitigation of consequences following design basis accidents may be damaged by a single exposure fire.

[1] *Exposure Fire.* An exposure fire is a fire in a given area that involves either in situ or transient combustibles and is external to any structures, systems, or components located in or adjacent to that same area. The effects of such fire (e.g., smoke, heat, or ignition) can adversely affect those structures, systems, or components important to safety. Thus, a fire involving one train of safe shutdown equipment may constitute an exposure fire for the redundant train located in the same area, and a fire involving combustibles other than either redundant train may constitute an exposure fire to both redundant trains located in the same area.

The most stringent fire damage limit shall apply for those systems that fall into more than one category. Redundant systems used to mitigate the consequences of other design basis accidents but not necessary for safe shutdown may be lost to a single exposure fire. However, protection shall be provided so that a fire within only one such system will not damage the redundant system.

II. General Requirements

A. *Fire protection program.* A fire protection program shall be established at each nuclear power plant. The program shall establish the fire protection policy for the protection of structures, systems, and components important to safety at each plant and the procedures, equipment, and personnel required to implement the program at the plant site.

The fire protection program shall be under the direction of an individual who has been delegated authority commensurate with the responsibilities of the position and who has available staff personnel knowledgeable in both fire protection and nuclear safety.

The fire protection program shall extend the concept of defense-in-depth to fire protection in fire areas important to safety, with the following objectives:

• To prevent fires from starting;

• To detect rapidly, control, and extinguish promptly those fires that do occur;

• To provide protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant.

B. *Fire hazards analysis.* A fire hazards analysis shall be performed by qualified fire protection and reactor systems engineers to (1) consider potential in situ and transient fire hazards; (2) determine the consequences of fire in any location in the plant on the ability to safely shut down the reactor or on the ability to minimize and control the release of radioactivity to the environment; and (3) specify measures for fire prevention, fire detection, fire suppression, and fire containment and alternative shutdown capability as required for each fire area containing structures, systems, and components important to safety in accordance with NRC guidelines and regulations.

C. *Fire prevention features.* Fire protection features shall meet the following general requirements for all fire areas that contain or present a fire hazard to structures, systems, or components important to safety.

1. In situ fire hazards shall be identified and suitable protection provided.

2. Transient fire hazards associated with normal operation, maintenance, repair, or modification activities shall be identified and eliminated where possible. Those transient fire hazards that can not be eliminated shall be controlled and suitable protection provided.

3. Fire detection systems, portable extinguishers, and standpipe and hose stations shall be installed.

4. Fire barriers or automatic suppression systems or both shall be installed as necessary to protect redundant systems or components necessary for safe shutdown.

5. A site fire brigade shall be established, trained, and equipped and shall be on site at all times.

6. Fire detection and suppression systems shall be designed, installed, maintained, and tested by personnel properly qualified by experience and training in fire protection systems.

7. Surveillance procedures shall be established to ensure that fire barriers are in place and that fire suppression systems and components are operable.

D. *Alternative or dedicated shutdown capability.* In areas where the fire protection features cannot ensure safe shutdown capability in the event of a fire in that area, alternative or dedicated safe shutdown capability shall be provided.

III. Specific Requirements

A. *Water supplies for fire suppression systems.* Two separate water supplies shall be provided to furnish necessary water volume and pressure to the fire main loop.

Each supply shall consist of a storage tank, pump, piping, and appropriate isolation and control valves. Two separate redundant suctions in one or more intake structures from a large body of water (river, lake, etc.) will satisfy the requirement for two separated water storage tanks. These supplies shall be separated so that a failure of one supply will not result in a failure of the other supply.

Each supply of the fire water distribution system shall be capable of providing for a period of 2 hours the maximum expected water demands as determined by the fire hazards analysis for safety-related areas or other areas that present a fire exposure hazard to safety-related areas.

When storage tanks are used for combined service-water/fire-water uses the minimum volume for fire uses shall be ensured by means of dedicated tanks or by some physical means such as a vertical standpipe for other water service. Administrative controls, including locks for tank outlet valves, are unacceptable as the only means to ensure minimum water volume.

Other water systems used as one of the two fire water supplies shall be permanently connected to the fire main system and shall be capable of automatic alignment to the fire main system. Pumps, controls, and power supplies in these systems shall satisfy the requirements for the main fire pumps. The use of other water systems for fire protection shall not be incompatible with their functions required for safe plant shutdown. Failure of the other system shall not degrade the fire main system.

B. *Sectional isolation valves.* Sectional isolation valves such as post indicator valves or key operated valves shall be installed in the fire main loop to permit isolation of portions of the fire main loop for maintenance or repair without interrupting the entire water supply.

C. *Hydrant isolation valves.* Valves shall be installed to permit isolation of outside hydrants from the fire main for maintenance or repair without interrupting the water supply to automatic or manual fire suppression systems in any area containing or presenting a fire hazard to safety-related or safe shutdown equipment.

D. *Manual fire suppression.* Standpipe and hose systems shall be installed so that at least one effective hose stream will be able to reach any location that contains or presents an exposure fire hazard to structures, systems, or components important to safety.

Access to permit effective functioning of the fire brigade shall be provided to all areas that contain or present an exposure fire hazard to structures, systems, or components important to safety.

Standpipe and hose stations shall be inside PWR containments and BWR containments that are not inerted. Standpipe and hose stations inside containment may be connected to a high quality water supply of sufficient quantity and pressure other than the fire main loop if plant-specific features prevent extending the fire main supply inside containment. For BWR drywells, standpipe and hose stations shall be placed outside the dry well with adequate lengths of hose to reach any location inside the dry well with an effective hose stream.

E. *Hydrostatic hose tests.* Fire hose shall be hydrostatically tested at a pressure of 150 psi or 50 psi above maximum fire main operating pressure, whichever is greater. Hose stored in outside hose houses shall be tested annually. Interior standpipe hose shall be tested every three years.

F. *Automatic fire detection.* Automatic fire detection systems shall be installed in all areas of the plant that contain or present an exposure fire hazard to safe shutdown or safety-related systems or components. These fire detection systems shall be capable of operating with or without offsite power.

G. *Fire protection of safe shutdown capability.* 1. Fire protection features shall be provided for structures, systems, and components important to safe shutdown. These features shall be capable of limiting fire damage so that:

a. One train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or emergency control station(s) is free of fire damage; and

b. Systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station(s) can be repaired within 72 hours.

2. Except as provided for in paragraph G.3 of this section, where cables or equipment, including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided:

a. Separation of cables and equipment and associated non-safety circuits of redundant trains by a fire barrier having a 3-hour rating. Structural steel forming a part of or supporting such fire barriers shall be protected to provide fire resistance equivalent to that required of the barrier;

b. Separation of cables and equipment and associated non-safety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustible or fire hazards. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area; or

c. Enclosure of cable and equipment and associated non-safety circuits of one redundant train in a fire barrier having a 1-hour rating, In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area;

Inside noninerted containments one of the fire protection means specified above or one of the following fire protection means shall be provided:

d. Separation of cables and equipment and associated non-safety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards;

e. Installation of fire detectors and an automatic fire suppression system in the fire area; or

f. Separation of cables and equipment and associated non-safety circuits of redundant trains by a noncombustible radiant energy shield.

3. Alternative of dedicated shutdown capability and its associated circuits,[1] independent of cables, systems or components in the area, room, zone under consideration should be provided:

a. Where the protection of systems whose function is required for hot shutdown does not satisfy the requirement of paragraph G.2 of this section; or

b. Where redundant trains of systems required for hot shutdown located in the same fire area may be subject to damage from fire suppression activities or from the rupture or inadvertent operation of fire suppression systems.

In addition, fire detection and a fixed fire suppression system shall be installed in the area, room, or zone under consideration.

H. *Fire brigade.* A site fire brigade trained and equipped for fire fighting shall be established to ensure adequate manual fire fighting capability for all areas of the plant containing structures, systems, or components important to safety. The fire brigade shall be at least five members on each shift. The brigade leader and at least two brigade members shall have sufficient training in or knowledge of plant safety-related systems to understand the effects of fire and fire suppressants on safe shutdown capability. The qualification of fire brigade members shall include an annual physical examination to determine their ability to perform strenuous fire fighting activities. The shift supervisor shall not be a member of the fire brigade. The brigade leader shall be competent to assess the potential safety consequences of a fire and advise control room personnel. Such competence by the brigade leader may be evidenced by possession of an operator's license or equivalent knowledge of plant safety-related systems.

The minimum equipment provided for the brigade shall consist of personal protective equipment such as turnout coats, boots, gloves, hard hats, emergency communications equipment, portable lights, portable ventilation equipment, and portable extinguishers. Self-contained breathing apparatus using full-face positive-pressure masks approved by NIOSH (National Institute for Occupational Safety and Health—approval formerly given by the U.S. Bureau of Mines) shall be provided for fire brigade, damage control, and control room personnel. At least 10 masks shall be available for fire brigade personnel. Control room personnel may be furnished breathing air by a manifold system piped from a storage reservoir if practical. Service or rated operating life shall be a minimum of one-half hour for the self-contained units.

At least a 1-hour supply of breathing air in extra bottles shall be located on the plant site for each unit of self-contained breathing appratus. In addition, an onsite 6-hour supply of reserve air shall be provided and arranged to permit quick and complete replenishment of exhausted air supply bottles as they are returned. If compressors are used as a source of breathing air, only units approved for breathing air shall be used and the compressors shall be operable assuming a loss of offsite power. Special care must be taken to locate the compressor in areas free of dust and contaminants.

I. *Fire brigade training.* The fire brigade training program shall ensure that the capability to fight potential fires is established and maintained. The program shall consist of an initial classroom instruction program followed by periodic classroom instruction, fire fighting practice, and fire drills:

1. *Instruction*

a. The initial classroom instruction shall include:

(1) Indoctrination of the plant fire fighting plan with specific identification of each individual's responsibilities.

(2) Identification of the type and location of fire hazards and associated types of fires that could occur in the plant.

(3) The toxic and corrosive characteristics of expected products of combustion.

(4) Identification of the location of fire fighting equipment for each fire area and familiarization with the layout of the plant, including access and egress routes to each area.

(5) The proper use of available fire fighting equipment and the correct method of fighting each type of fire. The types of fires covered should include fires in energized electrical equipment, fires in cables and cable trays, hydrogen fires, fires involving flammable and combustible liquids or hazardous process chemicals, fires resulting from construction or modifications (welding), and record file fires.

(6) The proper use of communication, lighting, ventilation, and emergency breathing equipment.

(7) The proper method for fighting fires inside buildings and confined spaces.

(8) The direction and coordination of the fire fighting activities (fire brigade leaders only).

(9) Detailed review of fire fighting strategies and procedures.

[1] Alternative shutdown capability is provided by rerouting, relocating, or modifying existing systems; dedicated shutdown capability is provided by installing new structures and systems for the function of postfire shutdown.

(10) Review of the latest plant modifications and corresponding changes in fire fighting plans.

NOTE: Items (9) and (10) may be deleted from the training of no more than two of the non-operations personnel who may be assigned to the fire brigade.

b. The instruction shall be provided by qualified individuals who are knowledgeable, experienced, and suitably trained in fighting the types of fires that could occur in the plant and in using the types of equipment available in the nuclear power plant.

c. Instruction shall be provided to all fire brigade members and fire brigade leaders.

d. Regular planned meetings shall be held at least every 3 months for all brigade members to review changes in the fire protection program and other subjects as necessary.

e. Periodic refresher training sessions shall be held to repeat the classroom instruction program for all brigade members over a two-year period. These sessions may be concurrent with the regular planned meetings.

2. *Practice*

Practice sessions shall be held for each shift fire brigade on the proper method of fighting the various types of fires that could occur in a nuclear power plant. These sessions shall provide brigade members with experience in actual fire extinguishment and the use of emergency breathing apparatus under strenuous conditions encountered in fire fighting. These practice sessions shall be provided at least once per year for each fire brigade member.

3. *Drills*

a. Fire brigade drills shall be performed in the plant so that the fire brigade can practice as a team.

b. Drills shall be performed at regular intervals not to exceed 3 months for each shift fire brigade. Each fire brigade member should participate in each drill, but must participate in at least two drills per year.

A sufficient number of these drills, but not less than one for each shift fire brigade per year, shall be unannounced to determine the fire fighting readiness of the plant fire brigade, brigade leader, and fire protection systems and equipment. Persons planning and authorizing an unannounced drill shall ensure that the responding shift fire brigade members are not aware that a drill is being planned until it is begun. Unannounced drills shall not be scheduled closer than four weeks.

At least one drill per year shall be performed on a ''back shift'' for each shift fire brigade.

c. The drills shall be preplanned to establish the training objectives of the drill and shall be critiqued to determine how well the training objectives have been met. Unannounced drills shall be planned and critiqued by members of the management staff responsible for plant safety and fire protection. Performance deficiencies of a fire brigade or of individual fire brigade members shall be remedied by scheduling additional training for the brigade or members. Unsatisfactory drill performance shall be followed by a repeat drill within 30 days.

d. At 3-year intervals, a randomly selected unannounced drill must be critiqued by qualified individuals independent of the licensee's staff. A copy of the written report from these individuals must be available for NRC review and shall be retained as a record as specified in section III.I.4 of this appendix.

e. Drills shall as a minimum include the following:

(1) Assessment of fire alarm effectiveness, time required to notify and assemble fire brigade, and selection, placement and use of equipment, and fire fighting strategies.

(2) Assessment of each brigade member's knowledge of his or her role in the fire fighting strategy for the area assumed to contain the fire. Assessment of the brigade member's conformance with established plant fire fighting procedures and use of fire fighting equipment, including self-contained emergency breathing apparatus, communication equipment, and ventilation equipment, to the extent practicable.

(3) The simulated use of fire fighting equipment required to cope with the situation and type of fire selected for the drill. The area and type of fire chosen for the drill should differ from those used in the previous drill so that brigade members are trained in fighting fires in various plant areas. The situation selected should simulate the size and arrangement of a fire that could reasonably occur in the area selected, allowing for fire development due to the time required to respond, to obtain equipment, and organize for the fire, assuming loss of automatic suppression capability.

(4) Assessment of brigade leader's direction of the fire fighting effort as to thoroughness, accuracy, and effectiveness.

4. *Records*

Individual records of training provided to each fire brigade member, including drill critiques, shall be maintained for at least 3 years to ensure that each member receives training in all parts of the training program. These records of training shall be available for NRC review. Retraining or broadened training for fire fighting within buildings shall be scheduled for all those brigade members whose performance records show deficiencies.

J. *Emergency lighting.* Emergency lighting units with at least an 8-hour battery power supply shall be provided in all areas needed for operation of safe shutdown equipment and in access and egress routes thereto.

K. *Administrative controls.* Administrative controls shall be established to minimize fire hazards in areas containing structures, systems, and components important to safety. These controls shall establish procedures to:

1. Govern the handling and limitation of the use of ordinary combustible materials, combustible and flammable gases and liquids, high efficiency particulate air and charcoal filters, dry ion exchange resins, or other combustible supplies in safety-related areas.

2. Prohibit the storage of combustibles in safety-related areas or establish designated storage areas with appropriate fire protection.

3. Govern the handling of and limit transient fire loads such as combustible and flammable liquids, wood and plastic products, or other combustible materials in buildings containing safety-related systems or equipment during all phases of operating, and especially during maintenance, modification, or refueling operations.

4. Designate the onsite staff member responsible for the inplant fire protection review of proposed work activities to identify potential transient fire hazards and specify required additional fire protection in the work activity procedure.

5. Govern the use of ignition sources by use of a flame permit system to control welding, flame cutting, brazing, or soldering operations. A separate permit shall be issued for each area where work is to be done. If work continues over more than one shift, the permit shall be valid for not more than 24 hours when the plant is operating or for the duration of a particular job during plant shutdown.

6. Control the removal from the area of all waste, debris, scrap, oil spills, or other combustibles resulting from the work activity immediately following completion of the activity, or at the end of each work shift, whichever comes first.

7. Maintain the periodic housekeeping inspections to ensure continued compliance with these administrative controls.

8. Control the use of specific combustibles in safety-related areas. All wood used in safety-related areas during maintenance, modification, or refueling operations (such as lay-down blocks or scaffolding) shall be treated with a flame retardant. Equipment or supplies (such as new fuel) shipped in untreated combustible packing containers may be unpacked in safety-related areas if required for valid operating reasons. However, all combustible materials shall be removed from the area immediately following the unpacking. Such transient combustible material, unless stored in approved containers, shall not be left unattended during lunch breaks, shift changes, or other similar periods. Loose combustible packing material such as wood or paper excelsior, or polyethylene sheeting shall be placed in metal containers with tight-fitting self-closing metal covers.

9. Control actions to be taken by an individual discovering a fire, for example, notification of control room, attempt to extinguish fire, and actuation of local fire suppression systems.

10. Control actions to be taken by the control room operator to determine the need for brigade assistance upon report of a fire or receipt of alarm on control room annunciator panel, for example, announcing location of fire over PA system, sounding fire alarms, and notifying the shift supervisor and the fire brigade leader of the type, size, and location of the fire.

11. Control actions to be taken by the fire brigade after notification by the control room operator of a fire, for example, assembling in a designated location, receiving directions from the fire brigade leader, and discharging specific fire fighting responsibilities including selection and transportation of fire fighting equipment to fire location, selection of protective equipment, operating instructions for use of fire suppression systems, and use of preplanned strategies for fighting fires in specific areas.

12. Define the strategies for fighting fires in all safety-related areas and areas presenting a hazard to safety-related equipment. These strategies shall designate:

a. Fire hazards in each area covered by the specific prefire plans.

b. Fire extinguishants best suited for controlling the fires associated with the fire hazards in that area and the nearest location of these extinguishants.

c. Most favorable direction from which to attack a fire in each area in view of the ventilation direction, access hallways, stairs, and doors that are most likely to be free of fire, and the best station or elevation for fighting the fire. All access and egress routes that involve locked doors should be specifically identified in the procedure with the appropriate precautions and methods for access specified.

d. Plant systems that should be managed to reduce the damage potential during a local fire and the location of local and remote controls for such management (e.g., any hydraulic or electrical systems in the zone covered by the specific fire fighting procedure that could increase the hazards in the area because of overpressurization or electrical hazards).

e. Vital heat-sensitive system components that need to be kept cool while fighting a local fire. Particularly hazardous combustibles that need cooling should be designated.

f. Organization of fire fighting brigades and the assignment of special duties according to job title so that all fire fighting functions are covered by any complete shift personnel complement. These duties include command control of the brigade, transporting fire suppression and support equipment to the fire scenes, applying the extinguishant to the fire, communication with the control room, and coordination with outside fire departments.

g. Potential radiological and toxic hazards in fire zones.

h. Ventilation system operation that ensures desired plant air distribution when the ventilation flow is modified for fire containment or smoke clearing operations.

i. Operations requiring control room and shift engineer coordination or authorization.

j. Instructions for plant operators and general plant personnel during fire.

L. *Alternative and dedicated shutdown capability*. 1. Alternative or dedicated shutdown capability provided for a specific fire area shall be able to (a) achieve and maintain subcritical reactivity conditions in the reactor; (b) maintain reactor coolant inventory; (c) achieve and maintain hot standby[2] conditions for a PWR (hot shutdown[2] for a BWR); (d) achieve cold shutdown conditions within 72 hours; and (e) maintain cold shutdown conditions thereafter. During the postfire shutdown, the reactor coolant system process variables shall be maintained within those predicted for a loss of normal a.c. power, and the fission product boundary integrity shall not be affected; i.e., there shall be no fuel clad damage, rupture of any primary coolant boundary, of rupture of the containment boundary.

2. The performance goals for the shutdown functions shall be:

a. The reactivity control function shall be capable of achieving and maintaining cold shutdown reactivity conditions.

b. The reactor coolant makeup function shall be capable of maintaining the reactor coolant level above the top of the core for BWRs and be within the level indication in the pressurizer for PWRs.

c. The reactor heat removal function shall be capable of achieving and maintaining decay heat removal.

d. The process monitoring function shall be capable of providing direct readings of the process variables necessary to perform and control the above functions.

e. The supporting functions shall be capable of providing the process cooling, lubrication, etc., necessary to permit the operation of the equipment used for safe shutdown functions.

3. The shutdown capability for specific fire areas may be unique for each such area, or it may be one unique combination of systems for all such areas. In either case, the alternative shutdown capability shall be independent of the specific fire area(s) and shall accommodate postfire conditions where offsite power is available and where offsite power is not available for 72 hours. Procedures shall be in effect to implement this capability.

4. If the capability to achieve and maintain cold shutdown will not be available because of fire damage, the equipment and systems comprising the means to achieve and maintain the hot standby or hot shutdown condition shall be capable of maintaining such conditions until cold shutdown can be achieved. If such equipment and systems will not be capable of being powered by both onsite and offsite electric power systems because of fire damage, an independent onsite power system shall be provided. The number of operating shift personnel, exclusive of fire brigade members, required to operate such equipment and systems shall be on site at all times.

5. Equipment and systems comprising the means to achieve and maintain cold shutdown conditions shall not be damaged by fire; or the fire damage to such equipment and systems shall be limited so that the systems can be made operable and cold shutdown can be achieved within 72 hours. Materials for such repairs shall be readily available on site and procedures shall be in effect to implement such repairs. If such equipment and systems used prior to 72 hours after the fire will not be capable of being powered by both onsite and offsite electric power systems because of fire damage, an independent onsite power system shall be provided. Equipment and systems used after 72 hours may be powered by offsite power only.

6. Shutdown systems installed to ensure postfire shutdown capability need not be designed to meet seismic Category I criteria, single failure criteria, or other design basis accident criteria, except where required for other reasons, e.g., because of interface with or impact on existing safety systems, or because of adverse valve actions due to fire damage.

7. The safe shutdown equipment and systems for each fire area shall be known to be isolated from associated non-safety circuits in the fire area so that hot shorts, open circuits, or shorts to ground in the associated circuits will not prevent operation of the safe shutdown equipment. The separation and barriers between trays and conduits containing associated circuits of one safe shutdown division and trays and conduits containing associated circuits or safe shutdown cables from the redundant division, or the isolation of these associated circuits from the safe shutdown equipment, shall be such

[2] As defined in the Standard Technical Specifications.

that a postulated fire involving associated circuits will not prevent safe shutdown.[3]

M. *Fire barrier cable penetration seal qualification.* Penetration seal designs must be qualified by tests that are comparable to tests used to rate fire barriers. The acceptance criteria for the test must include the following:

1. The cable fire barrier penetration seal has withstood the fire endurance test without passage of flame or ignition of cables on the unexposed side for a period of time equivalent to the fire resistance rating required of the barrier;

2. The temperature levels recorded for the unexposed side are analyzed and demonstrate that the maximum temperature is sufficiently below the cable insulation ignition temperature; and

3. The fire barrier penetration seal remains intact and does not allow projection of water beyond the unexposed surface during the hose stream test.

N. *Fire doors.* Fire doors shall be self-closing or provided with closing mechanisms and shall be inspected semiannually to verify that automatic hold-open, release, and closing mechanisms and latches are operable.

One of the following measures shall be provided to ensure they will protect the opening as required in case of fire:

1. Fire doors shall be kept closed and electrically supervised at a continuously manned location;

2. Fire doors shall be locked closed and inspected weekly to verify that the doors are in the closed position;

3. Fire doors shall be provided with automatic hold-open and release mechanisms and inspected daily to verify that doorways are free of obstructions; or

4. Fire doors shall be kept closed and inspected daily to verify that they are in the closed position.

The fire brigade leader shall have ready access to keys for any locked fire doors.

Areas protected by automatic total flooding gas suppression systems shall have electrically supervised self-closing fire doors or shall satisfy option 1 above.

O. *Oil collection system for reactor coolant pump.* The reactor coolant pump shall be equipped with an oil collection system if the containment is not inerted during normal operation. The oil collection system shall be so designed, engineered, and installed that failure will not lead to fire during normal or design basis accident conditions and that there is reasonable assurance that the system will withstand the Safe Shutdown Earthquake.[4]

Such collection systems shall be capable of collecting lube oil from all potential pressurized and unpressurized leakage sites in the reactor coolant pump lube oil systems. Leakage shall be collected and drained to a vented closed container that can hold the entire lube oil system inventory. A flame arrester is required in the vent if the flash point characteristics of the oil present the hazard of fire flashback. Leakage points to be protected shall include lift pump and piping, overflow lines, lube oil cooler, oil fill and drain lines and plugs, flanged connections on oil lines, and lube oil reservoirs where such features exist on the reactor coolant pumps. The drain line shall be large enough to accommodate the largest potential oil leak.

[45 FR 76611, Nov. 19, 1980; 46 FR 44735, Sept. 8, 1981, as amended at 53 FR 19251, May 27, 1988; 65 FR 38191, June 20, 2000]

APPENDIX S TO PART 50—EARTHQUAKE ENGINEERING CRITERIA FOR NUCLEAR POWER PLANTS

GENERAL INFORMATION

This appendix applies to applicants for a design certification or combined license pursuant to part 52 of this chapter or a construction permit or operating license pursuant to part 50 of this chapter on or after January 10, 1997. However, for either an operating license applicant or holder whose construction permit was issued prior to January 10, 1997, the earthquake engineering criteria in Section VI of appendix A to 10 CFR part 100 continues to apply.

I. INTRODUCTION

(a) Each applicant for a construction permit, operating license, design certification, or combined license is required by §50.34 (a)(12), (b)(10), and General Design Criterion 2 of appendix A to this part to design nuclear power plant structures, systems, and components important to safety to withstand the effects of natural phenomena, such as earthquakes, without loss of capability to perform their safety functions. Also, as specified in §50.54(ff), nuclear power plants that have implemented the earthquake engineering criteria described herein must shut down if the criteria in Paragraph IV(a)(3) of this appendix are exceeded.

(b) These criteria implement General Design Criterion 2 insofar as it requires structures, systems, and components important to safety to withstand the effects of earthquakes.

[3] An acceptable method of complying with this alternative would be to meet Regulatory Guide 1.75 position 4 related to associated circuits and IEEE Std 384–1974 (Section 4.5) where trays from redundant safety divisions are so protected that postulated fires affect trays from only one safety division.

[4] See Regulatory Guide 1.29—''Seismic Design Classification'' paragraph C.2.

II. SCOPE

The evaluations described in this appendix are within the scope of investigations permitted by § 50.10(c)(1).

III. DEFINITIONS

As used in these criteria:

Combined license means a combined construction permit and operating license with conditions for a nuclear power facility issued pursuant to subpart C of part 52 of this chapter.

Design Certification means a Commission approval, issued pursuant to subpart B of part 52 of this chapter, of a standard design for a nuclear power facility. A design so approved may be referred to as a ''certified standard design.''

The *Operating Basis Earthquake Ground Motion (OBE)* is the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public will remain functional. The Operating Basis Earthquake Ground Motion is only associated with plant shutdown and inspection unless specifically selected by the applicant as a design input.

A *response spectrum* is a plot of the maximum responses (acceleration, velocity, or displacement) of idealized single-degree-of-freedom oscillators as a function of the natural frequencies of the oscillators for a given damping value. The response spectrum is calculated for a specified vibratory motion input at the oscillators' supports.

The *Safe Shutdown Earthquake Ground Motion* (SSE) is the vibratory ground motion for which certain structures, systems, and components must be designed to remain functional.

The *structures, systems, and components required to withstand the effects of the Safe Shutdown Earthquake Ground Motion or surface deformation* are those necessary to assure:

(1) The integrity of the reactor coolant pressure boundary;

(2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or

(3) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of § 50.34(a)(1).

Surface deformation is distortion of geologic strata at or near the ground surface by the processes of folding or faulting as a result of various earth forces. Tectonic surface deformation is associated with earthquake processes.

IV. APPLICATION TO ENGINEERING DESIGN

The following are pursuant to the seismic and geologic design basis requirements of § 100.23 of this chapter:

(a) Vibratory Ground Motion.

(1) Safe Shutdown Earthquake Ground Motion.

(i) The Safe Shutdown Earthquake Ground Motion must be characterized by free-field ground motion response spectra at the free ground surface. In view of the limited data available on vibratory ground motions of strong earthquakes, it usually will be appropriate that the design response spectra be smoothed spectra. The horizontal component of the Safe Shutdown Earthquake Ground Motion in the free-field at the foundation level of the structures must be an appropriate response spectrum with a peak ground acceleration of at least 0.1g.

(ii) The nuclear power plant must be designed so that, if the Safe Shutdown Earthquake Ground Motion occurs, certain structures, systems, and components will remain functional and within applicable stress, strain, and deformation limits. In addition to seismic loads, applicable concurrent normal operating, functional, and accident-induced loads must be taken into account in the design of these safety-related structures, systems, and components. The design of the nuclear power plant must also take into account the possible effects of the Safe Shutdown Earthquake Ground Motion on the facility foundations by ground disruption, such as fissuring, lateral spreads, differential settlement, liquefaction, and landsliding, as required in § 100.23 of this chapter.

(iii) The required safety functions of structures, systems, and components must be assured during and after the vibratory ground motion associated with the Safe Shutdown Earthquake Ground Motion through design, testing, or qualification methods.

(iv) The evaluation must take into account soil-structure interaction effects and the expected duration of vibratory motion. It is permissible to design for strain limits in excess of yield strain in some of these safety-related structures, systems, and components during the Safe Shutdown Earthquake Ground Motion and under the postulated concurrent loads, provided the necessary safety functions are maintained.

(2) Operating Basis Earthquake Ground Motion.

(i) The Operating Basis Earthquake Ground Motion must be characterized by response spectra. The value of the Operating Basis Earthquake Ground Motion must be set to one of the following choices:

(A) One-third or less of the Safe Shutdown Earthquake Ground Motion design response spectra. The requirements associated with this Operating Basis Earthquake Ground Motion in Paragraph (a)(2)(i)(B)(*I*) can be satisfied without the applicant performing explicit response or design analyses, or

(B) A value greater than one-third of the Safe Shutdown Earthquake Ground Motion design response spectra. Analysis and design must be performed to demonstrate that the requirements associated with this Operating Basis Earthquake Ground Motion in Paragraph (a)(2)(i)(B)(*I*) are satisfied. The design must take into account soil-structure interaction effects and the duration of vibratory ground motion.

(*I*) When subjected to the effects of the Operating Basis Earthquake Ground Motion in combination with normal operating loads, all structures, systems, and components of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public must remain functional and within applicable stress, strain, and deformation limits.

(3) Required Plant Shutdown. If vibratory ground motion exceeding that of the Operating Basis Earthquake Ground Motion or if significant plant damage occurs, the licensee must shut down the nuclear power plant. If systems, structures, or components necessary for the safe shutdown of the nuclear power plant are not available after the occurrence of the Operating Basis Earthquake Ground Motion, the licensee must consult with the Commission and must propose a plan for the timely, safe shutdown of the nuclear power plant. Prior to resuming operations, the licensee must demonstrate to the Commission that no functional damage has occurred to those features necessary for continued operation without undue risk to the health and safety of the public and the licensing basis is maintained.

(4) Required Seismic Instrumentation. Suitable instrumentation must be provided so that the seismic response of nuclear power plant features important to safety can be evaluated promptly after an earthquake.

(b) Surface Deformation. The potential for surface deformation must be taken into account in the design of the nuclear power plant by providing reasonable assurance that in the event of deformation, certain structures, systems, and components will remain functional. In addition to surface deformation induced loads, the design of safety features must take into account seismic loads and applicable concurrent functional and accident-induced loads. The design provisions for surface deformation must be based on its postulated occurrence in any direction and azimuth and under any part of the nuclear power plant, unless evidence indicates this assumption is not appropriate, and must take into account the estimated rate at which the surface deformation may occur.

(c) Seismically Induced Floods and Water Waves and Other Design Conditions. Seismically induced floods and water waves from either locally or distantly generated seismic activity and other design conditions determined pursuant to §100.23 of this chapter must be taken into account in the design of the nuclear power plant so as to prevent undue risk to the health and safety of the public.

[61 FR 65173, Dec. 11, 1996]